NUCLEAR ENGINEERING VOL. 5 NO. 53 OCTOBER, 1960

The American Scene

AS the centre for the Allies' development of the atomic bomb, America had, by 1945, already established a wide network of research establishments and production organizations with considerable competence in the new science. Whilst in those early days the work was entirely concerned with the military aspects of atomic energy, both the technologies evolved and the actual plants were to have a great bearing on subsequent peaceful applications. The continuing requirement for fissile material for military stock piles was no less significant and the United States' civilian programme was able to come into being against a background of established expertize, and with a knowledge that the supply of enriched material would not be a problem. Even today, fifteen years later, America is unique in the Western World in having large stocks of enriched material available for civilian developments.

The price which has been established for U^{235} has itself had a material bearing on the approach to civilian reactor problems. Some years ago there was much criticism from competitive countries that this price was artificially low but, bearing in mind the very vast scale of the U.S. enrichment plants, the low cost of electrical power, and the revised cost estimates that have been leaking out of France and the United Kingdom, it has become clear that the price scale which the U.S. AEC established does not contain hidden subsidies.

Military Legacies

The availability of low cost enriched material and the existence of highly specialized National Laboratories and a wide range of industrial organizations experienced in nuclear technology were not the only legacies of the military requirements. The prime peace-time project became the establishment of a nuclear powered submarine fleet and at least the examination of the potentialities of nuclear propulsion for surface naval vessels. For this programme the rapid development of the most readily available and suitable technology was necessary, with the result that the pressurized water reactor (with the sodium cooled reactor as a losing competitor) became the spear-head of the power reactor development programme. So dynamic was the naval drive that even the design of Shippingport, the first large-scale land-based reactor in the United States, was heavily angled towards the technology of naval ship propulsion.

In view of the early start and the heavy expenditure on development, it may appear strange that only now are numbers of large-scale plants being brought into commission. But apart from the naval projects, the early developments in the United States were on too wide a front for the immediate objectives to be clearly defined, save the very general one of developing reactor systems which looked economically attractive. In fact, a criticism frequently levelled at the United States by outside countries (possibly also tinged with envy) was that so broad was the enterprise and so large was the amount of money available, no effective co-ordination or leadership could be maintained. Much of the effort was frittered away in duplication of work and in the examination of high-flown ideas whose impracticability was evident from the outset. Little accent was placed upon the construction of plants or on the engineering implications of full scale operation, but a great deal of money was spent in paper studies. At one time it appeared that good publicity for an idea was more important than technical merit.

AEC-Industry Relationships

The reactor building programme has also been hampered by the anxiety of the AEC to steer clear of any position in which it could be accused of treading upon the preserves of the utility companies. A driving force throughout, and one which goes far to explain the present relationship between the AEC and industry, is the requirement that industry should be brought into any project at as early a stage as possible and all industrial operations should eventually be taken over by industry preferably on a commercial basis but otherwise on a contract basis. Government interference in industrial operations is, of course, anathema to American society. This has resulted in such AEC establishments as the National Reactor Testing Station at Idaho Falls and the research and production facilities at Oak Ridge being operated by companies under contract. Much of the AEC's research too, is handled by industry under contract. Whilst this may mean a considerable turnover for industry, the system does carry disadvantages. AEC sponsored work is made freely available to all, without payment of royalties. Although it is not certain that all work for which the AEC pays is fully reported back nor that all patents evolving from AEC sponsored work are handed over, companies working on projects which they believe have a high commercial value are reluctant to undertake AEC contracts in order to safeguard their own patent position.

The determination to hand over to industry responsibility for so many activities has not resulted either in there being an easy profit. Indeed, under the policy of Admiral Strauss cut-throat competition was so actively encouraged, that at times one felt that the AEC would deliberately foster the establishment of conditions whereby a company was forced out of the nuclear field simply to prove the realism of the commercial competition. The oft-repeated phrase " what's

good for General Motors is good for the Atomic Energy Industry" gave more than a hint of early AEC policy.

The real and imagined competition between Government-owned and privately owned generating authorities has also done much to bedevil the reactor construction programme and resulted in so many arguments of technical importance being decided on a political basis. The joint Senate-Congress committee on atomic energy can also be accused of hindering developments by direct intervention in the AEC's programmes, for example, in relation to gas cooled reactors.

Since the accession of Mr. McCone to the position of chairman of the AEC, matters have improved considerably as he carried the confidence of both major political parties and industry when he took over the post. Over the past two years there has been a progressive smoothing of the relations between the Government, the AEC and industry and a steady rationalization of the programme to concentrate on the more profitable projects. Whilst the number of main-line projects has diminished the American programme is still broadly based but the major problem of organizing a continuing construction programme has still to be resolved.

So many of the difficulties would never have arisen if there had been at the outset a genuine home market for nuclear reactors in the United States. But fossil fuel is cheaper there than almost anywhere else in the world and, even in the high cost areas, generating costs are relatively low compared with the bulk of industrialized countries. Not only would it have been easier to establish the short range technical objectives, so also would many of the legal and political problems have been simplified. In spite of the number of atomic energy installations and research reactors in the country, the establishment of indemnity regulations has been regrettably slow. Wrangling over this subject can also be held responsible, at least in part, for the public relations problem that is still much more acute in the United States than in Europe. It is difficult to assess fully the retarding effect of public antipathy or its influence on development and production costs but it certainly cannot be ignored.

Industry itself is, of course, responsible for some of its present difficulties. As in so many countries atomic energy was regarded as an industrial band wagon; much irresponsible cut-price quoting occurred and the years of over-optimism led to heavy financial burdens. Even so, the AEC has not been guiltless for its early encouragement of industry and its continuing exaggeration of potential. In the field of isotopes, for example, the claims that have been made for the savings in industrial plant costs that isotope techniques can produce must be regarded as extravagantly optimistic.

A further factor leading to the creation of an unbalanced situation has been the desire for "leadership" in atomic energy of almost pathological intensity. This is, of course, a political objective more than a commercial one. On the technical floor and in the laboratories a much more down-to-earth and sober attitude is encountered. For a long time atomic energy for peaceful purposes suffered from being the main prestige banner of President Eisenhower and was used as a political weapon (albeit for peaceful collaboration). In such an atmosphere it is difficult to retain the more unemotional commercial approach to changing situations. Other nations will be the gainers as, for example, from the construction of the *Savannah*. This is an expensive project whose technical value is overshadowed by its function as the vehicle for defining, determining and resolving the rules and regulations that must be established before nuclear vessels can regularly ply the high seas.

If the political ambitions had been less vigorous, it is possible that negotiations with overseas countries would have been commercially more successful. Although the AEC is accused of subsidizing industry in regard to business overseas (and this is quite straightforward for research reactors) it frequently creates impossible commercial situations because of the political strings that are attached to, say, fuel agreements. The Euratom agreement demonstrates how the vital issues in selling power reactors abroad can be misinterpreted although, to be fair, Euratom contributed in large measure to the illusion. A simpler, direct subsidy for commercial construction would have been much more effective in inducing European utilities to co-operate, although no doubt, there would have been massive political pressures both at home and overseas opposing such a move.

In spite of the confusion of the past, there is steadily emerging a multipronged direct attack on the technical problems facing nuclear engineers. With so many reactor concepts entering the major operational phase, headed by Dresden and Yankee, the next 18 months will technically be highly exciting. The industrial difficulties immediately ahead will be partly assisted by the special programmes now gathering momentum, such as the space vehicle and remote site reactor programmes, but following the election of a new President next month the all-important problem of how to finance future full scale power plants must be resolved if an industrial hiatus is not to develop.

PRINCIPAL U.S. AEC R & D INSTALLATIONS

Argonne National Laboratory, Lemont, Illinois (University of Chicago).

Bettis Atomic Power Laboratory, Pittsburgh, Pa. (Westinghouse Electric Corp.).

Brookhaven National Laboratory, Upton, Long Island, N.Y. (Associated Universities Inc.).

Knolls Atomic Power Laboratory, Schenectady, N.Y. (General Electric Co.).

Los Alamos Scientific Laboratory, Los Alamos, N. Mex. (University of California).

Oak Ridge National Laboratory, Oak Ridge, Tenn. (Union Carbide Nuclear Co., Division of Union Carbide Corp.).

Raw Materials Development Laboratory, Winchester, Mass. (National Lead Co.).

University of California at Los Angeles, Atomic Energy Project, Los Angeles, Calif. (University of California).

National Reactor Testing Station (NRTS) Idaho Falls, Idaho, (Phillips Petroleum, Combustion Engineering, Aerojet General).

MAJOR AMERICAN POWER REACTOR MFRs.

Alco Products Inc., Schenectady 5, New York.

Allis-Chalmers Mfg. Co., Milwaukee 1, Wisconsin.

AMF Atomics (Division of American Machine & Foundry Co.), 140 Greenwich Avenue, Greenwich, Connecticut.

Atomics International (Division of North American Aviation Inc.), P.O. Box 309, Canoga Park, Calif.

Babcock & Wilcox Co., The, 161 East 42nd Street, New York 17, N.Y.

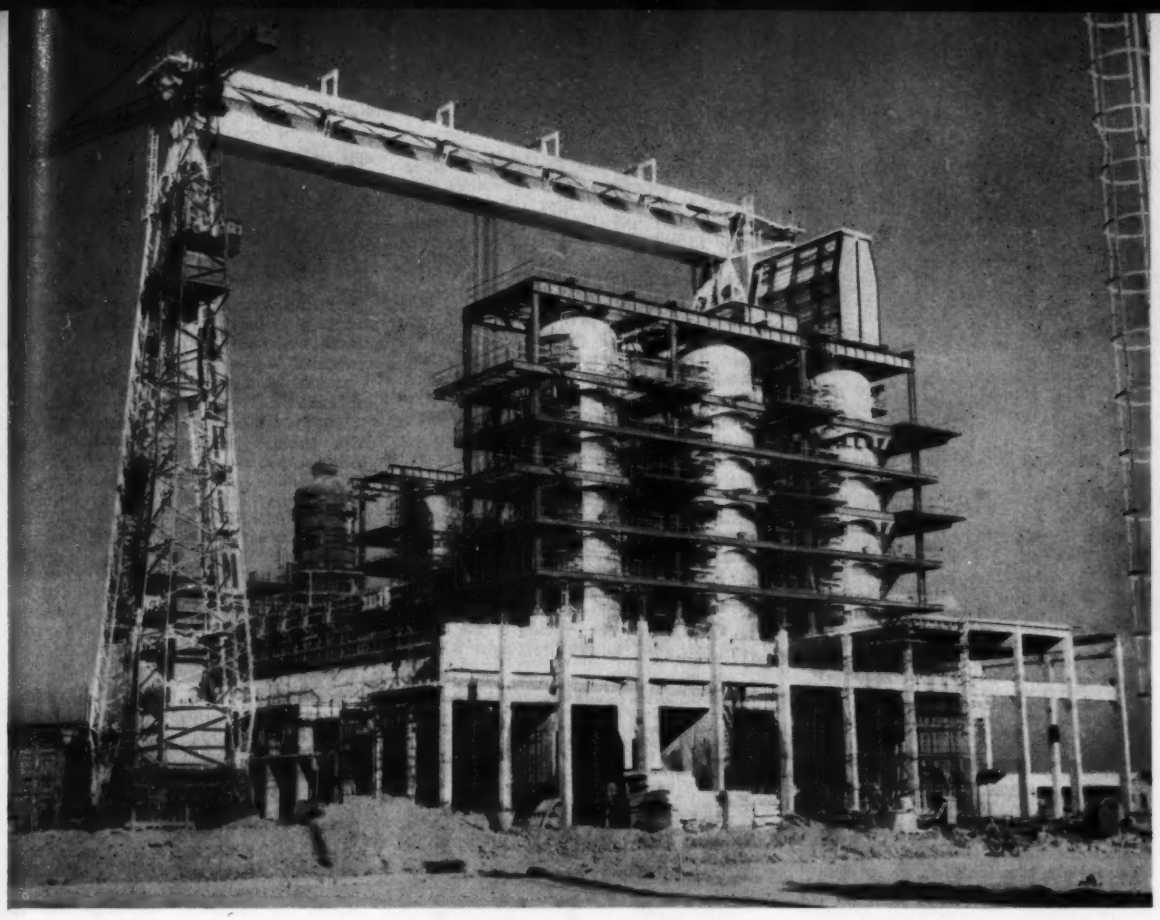
Combustion Engineering Inc., Nuclear Division, Windsor, Connecticut.

General Atomic (Division of General Dynamics Corporation), P.O. Box 608, San Diego 12, Calif.

General Electric Company, Atomic Power Equipment Dept., 2151 South First Street, San Jose 12, Calif.

Martin Company, The, Nuclear Division, Baltimore 3, Maryland.

Westinghouse Electric Corporation, Atomic Power Department, Engineering Center, P.O. Box 355, Pittsburgh 30, Pa.



World News

Work is proceeding smoothly at Latina where the Nuclear Power Group (previously NPPC) are building a 200 MW station for Societa Italiana Merdionale per l'Energia Atomica (SIMEA). As the latest photograph shows the fifth boiler is now in position. (Below) Installation of the top dome.

International

NUCLEAR POWER COSTS are the subject of an IAEA report recently published, particularly of moderate and large plants of established technology. Although the report notes that engineering studies on smaller reactors are being made in many countries it concludes that insufficient data is available for reliable cost estimates to be made.

LEGAL AND ADMINISTRATIVE problems of protection in the peaceful uses of atomic energy were discussed at an international symposium organized by Euratom in Brussels from September 5 to 8. Participants were not confined to the Messina countries but included delegates from the United Kingdom and the United States.

TRANSPORT REGULATIONS governing the movement of large radiation sources, low activity isotopes and fissionable material have been approved by the IAEA's Board of Governors.

United Kingdom

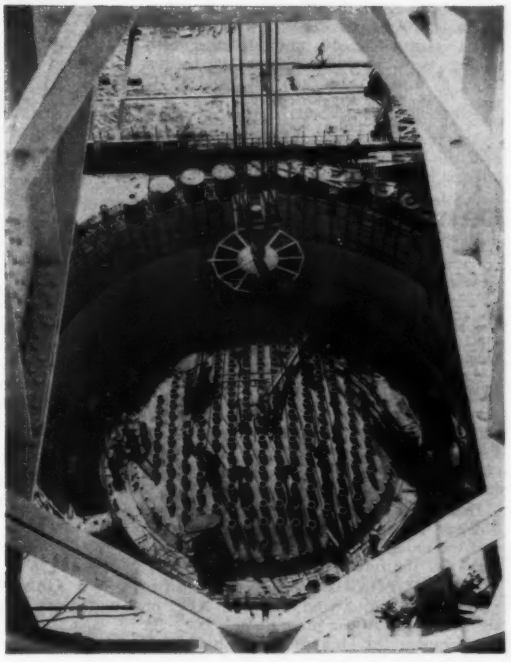
JASON, Hawker Siddeley's own Argonaut-type reactor at their works at Langley, which began operation towards the end of 1959, has been brought up to its full design power of 10 kW. The reactor has lately been used to provide design data for the installation based on this model at Winfrith, NESTOR. A 100 kW version of the reactor is under development.

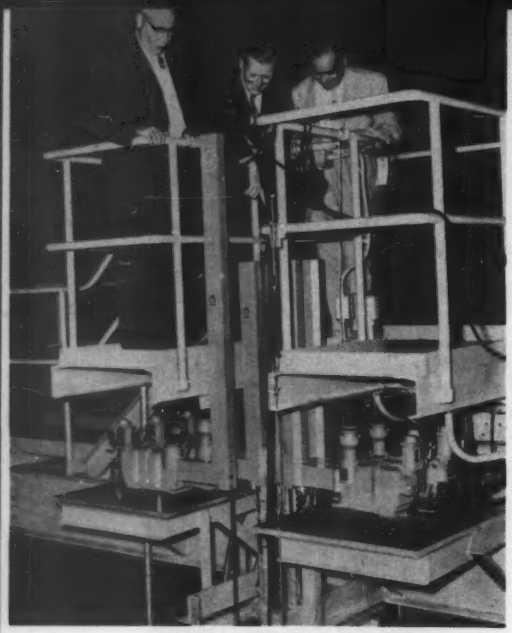
TRADE UNION officials attended a four-day course at the AEA's Isotope School at Wantage from September 20 to 23 on the industrial uses of radioisotopes. The programme included lectures, discussions and visits.

VISITORS to Winfrith during the Open Days (September 15 to 17) were shown five reactors in course of construction together with expotential experiments and sub-critical assemblies.

HECTOR—a new zero power reactor for studying nuclear characteristics of different power reactor designs making use of a "pile oscillator" technique—is to be built by Fairey Engineering at Winfrith. H. M. Hobson will produce the control gear for HECTOR.

DIMPLE — the heavy water critical assembly is to follow NERO and be moved from Harwell to Winfrith. Its control system will be redesigned.





Looking into the pool of the new $1,130,000 nuclear research reactor designed and installed by AMF's Atomic division at the University of Puerto Rico at Mayaguez, are, from left, John I. Thomas, Puerto Rico area manager of the U.S. AEC, William E. Chamberlain, vice president of AMF Atomics, and Sam R. Sapirie, manager, Oak Ridge Operations Office of the Commission. The reactor is the heart of a complete $3,500,000 nuclear research centre operated by the University on behalf of the AEC for the training of students and scientists from all of the American republics.

Australia

A DRIVE ON ISOTOPES is being made by the Australian Atomic Energy Commission, and companies and industrial concerns are to be encouraged to join in research and to send specialists to Lucas Heights in order to exploit known methods and evolve new techniques for using radiation in industry.

MOATA, aboriginal for an instrument for making fire, is the name that has been given to the UTR-10, the Argonaut-type reactor developed by Advanced Technology Laboratories scheduled to be in operation at Lucas Heights early in 1961. With a thermal power of 10 kW and a neutron flux of about 10^{11} n/cm² sec, the reactor comprises essentially a 4-ft graphite cube, containing two aluminium boxes (in section 20 in. by 6 in.) facing each other and 18 in. apart, each box being divided into six sections and containing 12 plates of aluminium clad uranium aluminium alloy, 26 in. x 3 in. x 0.080 in. Control is by four blades inserted in the core.

Canada

A DIRECT SALE of fuel elements will be made to the Japan Atomic Energy Research Institute by AMF Atomics Canada Limited covering 270 natural uranium elements for use in the JRR-3 research reactor—a 10-MW heavy water moderated reactor operated by JAERI. Part of the raw material was donated by Canada to the IAEA and then sold to Japan. The remainder of the metal was refined in Japan and shipped to AMF Atomics for fabrication. Previous sales of the company to foreign customers have always been through Atomic Energy of Canada Ltd.

INCORPORATION of the new Canadian Nuclear Association has been applied for, and the initial board of directors is studying plans for the future.

A JOINT BOARD is being established to advise A.E.C. Ltd. and the U.S. AEC on their co-operative programme for the development of heavy water moderated power reactors. The co-operation agreement between the two countries was amended in June to allow a fuller integration of effort; the U.S. is prepared to spend up to $5 million over the next five years in research specifically directed towards the Canadian reactors.

AN ARTIC PLANT generating 4.1 MW electricity and 35.9 MW of heat would cost 2% more to run if the plant were nuclear than if it were conventional oil burning, according to a report of Canadian Westinghouse. For a smaller size plant of 2 MWe and 17.5 MWt the disparity would be larger.

Denmark

A CHEQUE for $350,000 has been presented by the Government of the United States to the Government of Denmark towards the cost of a research reactor constructed for the Danish Atomic Energy Commission. Denmark is the fifth country to receive such a grant although the U.S. has made commitments to 20 countries. In Denmark's case the reactor is the 5-MW tank type DR-2 supplied by Foster Wheeler.

Canada is supplying 15,000 lb of uranium metal billets to Japan Atomic Energy Research Institute for JRR-3, a 10 MW heavy water research reactor. Shown here are George I. Staber, general manager, AMF Atomics (Canada); M. Sanda, president, Marubini-Jida (Canada) and H. S. Milne, AMF works manager.

France

A SECOND processing plant is to be built at La Hague, near Cherbourg, by the Commissiariat a l'Energie Atomique to supplement the existing plutonium extraction plant at Marcoule.

West Germany

AEG-ARGONAUT reactor, parts of which were featured at Hanover Trade Fair, has now been erected at Groz-Welzheim near Kahl and is ready for operation.

REACTOR at Kahl was due for criticality on September 15, with an initial loading of 88 elements.

1 MW swimming pool reactor at Munich is to be uprated to 2 MW by installing new elements and increasing heat exchange capacity.

India

A GAMMA GARDEN was inaugurated at New Delhi on August 25 to study the effect of radiation upon plants. A remotely controlled cobalt-60 source supplied by Canada is used to irradiate growing plants and the effects will be studied by the Indian Agricultural Research Institute.

Japan

JAERI has awarded General Electric Japan (a wholly owned affiliate of the American company, General Electric) a contract to build a 12.5 MW demonstration BWR at Tokai Mura. A letter of intent was sent to G.E. last year.

Puerto Rico

DEDICATION of the Nuclear Centre's new research reactor, a 1-MW pool type designed and installed by AMF at a cost of $1.13 million, was held on August 23. The reactor first went critical on August 16.

GROUND BREAKING for the BONUS BWR with superheat at Punta Higuera was also organized for August 23. $11 million has been authorized by Congress to be spent on this project for which the main design is being handled by G.N.E.C. Completion is scheduled for late 1962.

Spain

FINANCE to the extent of 18 million pesetas has been agreed by the Spanish Government for Spain's participation in the European Nuclear Energy Agency. Although not a full-time member of ENEA, Spain is a partner in the Eurochemic project.

Sweden

A VAN DE GRAAFF accelerator with a maximum power of 5.5 MV has been installed at AB Atom Energi's research station at Studsvik by High Voltage Engineering at a cost of $500,000.

U.S.A.

ENTERPRISE, an 85,000 ton U.S. aircraft carrier was scheduled for launching on September 24. Powered by eight PWR's, the vessel is due for commissioning in 1962.

SRE's second core is of thorium-uranium alloy. Each element comprises a 5-rod metallic duster containing 7.6 weight% uranium in thorium. Criticality was reached with 31 elements. An AI facility, SRE is located at Santa Susana near Los Angeles.

YANKEE Atomic Electric Company's 134 MW reactor (featured extensively in this issue) achieved criticality on August 19. Over the latter part of this year it will be progressively brought up to full power operation.

DEDICATION of the 180 MW Dresden station takes place on October 12. John A. McCone, chairman U.S. AEC, will be the principal speaker.

GCRE-1, the Gas Cooled Reactor Experiment which forms part of Aerojet General's closed cycle gas turbine reactor development programme was run up to its full power of 1,850 kW at NRTS for 48 hours before being shut down for examination. A prototype reactor of this type for the Department of Defence, the Mobile Low Power Nuclear Power Plant No. 1 (ML-1) is scheduled for operation next year. ML-1 is to be transported in four packets with a total weight less than 38 tons and is to be capable of operation within 12 hours of delivery to site.

HYDRAZINE PRODUCTION by fissiochemistry, using liquid ammonia as raw material, is the subject of a contract between Aerojet General and the Air Force Manufacturing Technology Division. Test equipment is being completed and in-pile experiments are to begin shortly.

THE THIRD POLARIS missile-firing submarine, *Robert E. Lee* has successfully completed its sea trials. The submarine, like its sister ships *George Washington* and *Patrick Henry*, can carry 18 solid-fuel Polaris missiles in vertical tubes from which they can be launched when the ship is submerged.

DIRECT CONVERSION using a cerium cell has produced 90 W direct from fission heating in an experiment at General Atomic's John Jay Hopkins Laboratory. The experiment was conducted in a Triga reactor with a fuel element of uranium carbide and zirconium carbide. At an operating temperature of 3,500°F an efficiency of 10% was achieved.

A TRIGA MARK II similar to the installation at Casaccia in Italy but with special experimental and irradiation facilities has begun operating at the University of Illinois Nuclear Reactor Laboratory, where it is licensed to operate at 100 kW.

CONTINUING OPERATION contracts have been awarded by the U.S. AEC to Goodyear Atomic Corp. and Westinghouse for the operation of the $760 million Portsmouth gaseous diffusion plant, Ohio, and the Bettis Atomic Power Laboratory in Pittsburgh, respectively. The Westinghouse contract is estimated to be valued at $50 million for the three-year period covered. National Lead has also been given a four-year extension of its contract to operate the $110 million feed materials production centre at Fernald, Ohio.

PHILLIPS PETROLEUM have been assigned the responsibility for operating the 20-40 MW experimental organic cooled reactor under construction at the NRTS. The facility, for which Congress appropriated $6 million, is scheduled to become critical in late 1962 when it will be used for general experiments on organic coolants.

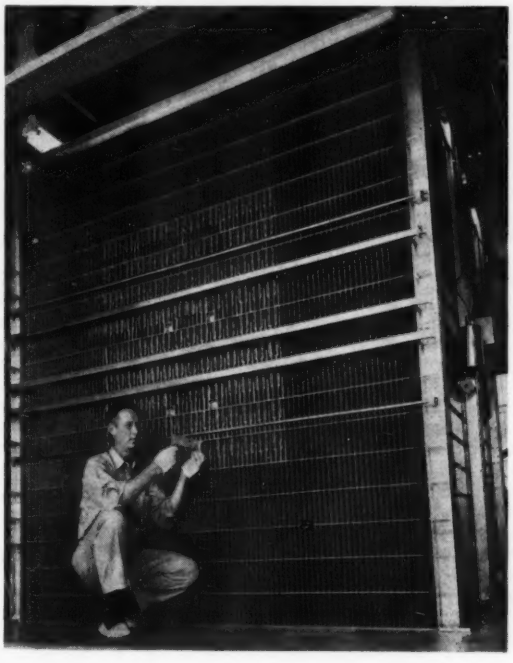
FUEL ELEMENTS for the new research reactor at the U.S. AEC's Puerto Rico nuclear centre were of a new type developed by ORNL. They comprised a dispersion of 20% enriched uranium oxide in aluminium, clad in aluminium, and they are expected to exhibit greater corrosion resistance because of an adherent alodized coating and the use of mechanical jointing instead of brazing.

AEC AID to non-profit institutions for nuclear education is to be extended to mid-1964 in order to boost education in nuclear technology. No further grants, however, will be made for the construction of teaching reactors. The future programme will be reorientated towards general education rather than reactor training.

MATERIALS RESEARCH CENTRE has been established at North Western University, Evanston, Illinois, sponsored by the Advanced Research Projects Agency. The object is to create a number of centres where materials research will be the responsibility of workers with a variety of discipline backgrounds.

LICENCE EXEMPTION has been given to certain classes of products containing small residual amounts of radio-activity such as petroleum which has contained traces for quality control. The AEC is also proposing to amend its regulations governing special nuclear materials to require the filing of reports covering individual movements of such materials.

DISPOSAL CONTAINERS for use in sea dumping are the subject of a proposed regulatory amendment which would define design criteria. Under this, containers would be required to retain integrity until they reach at least 1,000 fathoms depth, although in certain cases of low activity dense material a disposal container may be deemed unnecessary.



A versatile critical facility, at General Atomics' John Jay Hopkins Laboratory, for studying BeO-moderated cores for MGCR application. The 9-ft-cube honeycomb accepts fuel in the form of highly enriched uranium-aluminium foils and a central region simulated fuel elements and control rods. Initial experiments are being conducted with graphite moderation.

THE NORTH - WEST passage between the Atlantic and Pacific Oceans has been successfully navigated by the U.S. submarine *Seadragon*.

FUEL ELEMENTS for the new production reactor (NPR) at Hanford are to be manufactured by the AEC as quotations from industry exceeded by a substantial margin the AEC's own cost figures.

The core of SNAP-10 which first went critical on Aug. 2. The reactor, built by Atomics International, comprises a homogeneous mixture of zirconium hydride and uranium-235 reflected by beryllium; the whole unit weighs less than 200 lb.

A DISPOSAL LICENCE will be issued to Nuclear-Chem Disposal Corporation of Long Island (unless any forthcoming objections are sustained) to collect pre-packaged isotope material and source material for storage in an industrial area of New York for subsequent shipment to Oak Ridge. Proposed limitation of quantity is 1000 curies of tritium and 100 curies of other material.

THE OSMIUM ISOTOPES in gram quantities have been separated by the E-M separator at ORNL. The operation is complicated by the extreme toxity of the feed material, osmium oxide.

X-10 the oldest operating reactor which went critical in November, 1943, at ORNL, and uses graphite as a moderator, has been successfully annealed. For this purpose the flow of cooling air through the reactor was reversed.

RUTHENIUM contaminated with the 106 isotope which has been found in commercial jewellery is not regarded as representing a significant health hazard. Stocks of ruthenium held in the jewellery industry must, however, be disposed of.

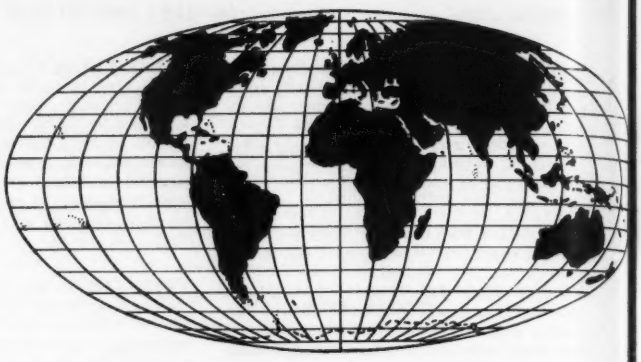
THE PRICE OF C^{14} has been reduced by ORNL by 27%, following the 50% price reduction 12 months ago. From July 1 the new rate of $9.5/mc has been charged. Since 1946 nearly 100,000 mc have been dispensed.

BERYLLIUM ORE deposits in Utah are to be exploited by Vitro Minerals, assisted by the first loan of the Office of Minerals Exploration relating to this element. The ore is of a new type and has been named Vitroite.

NUCLEAR REACTOR CHEMISTRY and Analytical Chemistry in Nuclear Reactor Technology will be the subject of a joint conference sponsored by Oak Ridge at Gatlinberg on October 12, 13 and 14. Proceedings will be published by the U.S. AEC.

THE JOURNAL of Applied Physics published for 31 years by the American Institute of Physics is to move to ORNL with Dr. Crawford as Editor.

The U.S. Civilian Power Reactor Programme



by FRANK K. PITTMAN
(Director, Division of Reactor Development U.S. AEC)

UNITED STATES Government policy in the development of civilian power reactors is guided mainly by established short-range and long-range economic objectives. The short-range objective is to make it possible between now and 1968 for utility owners in areas where fossil fuel costs are 35 cents per million B.t.u. or higher to select large sized nuclear plants over large sized conventional plants on the basis of economics alone.

The longer-range objective is to make it possible for essentially all utility owners to make such a choice regardless of location or size of plant. We have not established a fixed target date for this objective, but expect that during the seventies ever increasing numbers of utility owners will be able to choose nuclear power on the basis of economics.

This dual goal was created because we are convinced, first, that it is important to get competitive plants into being at an early date in order to lay a base for a healthy, publicly accepted nuclear power industry and, second, that very broad efforts are needed before such an industry can be firmly established. The United States Government and industry are pursuing a broadly-based development programme costing some $200 million per year and encompassing numerous sizes and types of nuclear systems offering a promise of meeting these objectives.

Our effort follows a three-phase sequence for the development of individual reactor concepts. This sequence comprises:

(1) A phase in which research in physics, engineering, and other scientific and technical work is undertaken out of pile, in critical experiments, reactor loops, and experimental reactors.

(2) Construction of prototypes to demonstrate certain principles on a commercial scale.

(3) Construction of plants of commercial proportions.

We believe that the facilities required for the second phase should generally be built and operated by publicly or privately owned utilities independently or in co-operation with the Commission, but, if this is not possible and the prototypes are necessary parts of the Commission programmes, they should be built and operated by the Commission. We, of course, expect industry to assume the primary responsibility for construction of commercial nuclear power stations.

We believe that the prototype plants should be integrated into utility power grids in order to demonstrate load factors, dependability, and other operating characteristics which affect the economics of the plants. In the development, construction, and operation of prototype and of commercial power producing reactors, the Commission will work co-operatively with reactor manufacturers and with publicly and privately owned utilities, and will, in certain instances, be willing to consider such assistance as research and development and fuel use charge waivers.

During the past few years, we have demonstrated that it is technically feasible to use several different reactor systems for the generation of electric power. Our evaluation of the economic potential of the several systems now under development leads us to conclude that it should be possible to achieve economically competitive nuclear power in high fuel cost areas of the United States between now and 1968 if we continue an aggressive programme of development of light water and organic cooled reactors.

Short-Range Objective

Reactors which are cooled and moderated with light water, fuelled with slightly enriched uranium in the form of UO_2 clad in stainless steel or zirconium, and which produce saturated steam for the turbines, have the greatest background of developed technology in the United States and are closest to attaining economic nuclear power. Our studies indicate that water reactor technology has already reached a point where, with reasonable extension, it should attain our short-range objective. The potential economic improvement offered by developments such as nuclear superheat, general engineering simplifications, and cheaper, longer-lived fuel, although not yet completely determined, should be sufficient to make the water reactors at least meet our short-range objective.

The organic cooled system is similar in many respects to the water system, but offers the additional potential advantages of operating at low pressure and of having fewer corrosion problems. These advantages are somewhat offset in that the organic materials now used are polymerized by radiation and must be continuously replaced, and heat transfer properties are not as good as those in water cooled systems. If the development programmes now under way are successful, however, and if fouling of heat transfer surfaces proves not to be a problem, there is every reason to believe that this system will also be capable of meeting our short-range objectives.

Longer-Range Objectives

Our long-range development programme includes the examination of several reactor systems which offer potential technical advantages over the water and organic cooled systems. In short, we are looking to systems which:

(1) Give higher temperatures.

(2) Offer higher neutron economy.

(3) Can be built at a more reasonable unit cost.

(4) Have lower fuel cycle, operation or maintenance costs.

It is quite likely that no one reactor system will offer all of these virtues and that the best systems of the future will be those in which there is a consolidation of the various desirable features. Our present knowledge indicates that the high temperature liquid metal and gas cooled systems, and particularly those reactors that show a high conversion ratio, have the greatest potential for meeting our long-range goals. It must be recognized, however, that these systems have many inherent technical difficulties that must be solved before their potential can be realized, and that the solution of these problems will be costly in time and money.

The part to be played by heavy water moderated natural uranium fuelled reactors in either the short or long range is uncertain and sensitive to such factors as methods of financing and terms of availability of fuel material and heavy water. The advantages of a natural uranium fuelled, heavy water moderated and cooled system are apparent. It eliminates dependence upon isotope separation plants in the fuel cycle, has high neutron economy, and requires relatively small quantities of uranium ore per unit of power produced. Our research and development effort will be directed toward the problems most critical to the evaluation of this system and, in this regard, we expect that there will be major mutual benefit from our co-operative programme with Canada.

We shall continue to study various fluid fuel systems, such as the aqueous homogeneous, molten salt, and molten plutonium concepts, as they offer potential fuel cycle savings which may well overcome the disadvantages inherent in handling large quantities of radioactive materials in fluid form.

We shall also investigate new reactor concepts or modifications to presently developed concepts. If engineering studies, research and development, and evaluation of these new ideas indicate that a concept is worth while, we shall, in general, build a reactor experiment to give us the information needed to determine the extent to which the concept can help meet our objectives.

We estimate that to carry out our planned programme for the 1960s, a total expenditure of about $2,000 million will be required. This cost forecast covers research and development, design and construction of experiments and prototypes, operation costs for the prototypes, and general engineering support of the civilian programme. The bulk of the expenditure undoubtedly will be made by the Government, but we are confident that industrial contribution will be at a substantial and ever increasing amount.

On the basis of nuclear plants now authorized for construction in the United States, we can expect a generating capacity of almost 1,400 MWe to be in operation by the end of 1964. Most of this will come into operation in the 1960-1962 period, during which time 17 nuclear plants are expected to go critical. These include a large fraction of the demonstration facilities which now appear to be required to determine whether or not our short-range objective can be met, as well as the experimental gas and liquid metal cooled facilities which will be needed for the longer range. The rate of prototype construction for the next year or so will be somewhat reduced in comparison with recent years. We shall continue to have a programme of reactor experiments and, hopefully, an industrial effort directed at construction of commercial reactors which will be very nearly or fully competitive with conventional plants in the areas where they are constructed.

Industrial Viewpoint

by **DR. CHAUNCEY STARR**

(President, Atomics International Division of North American Aviation, Inc.)

EVALUATIONS of the future of nuclear power in the United States by both technical and economic analysts indicate a feeling of cautious optimism for the years ahead. The hard-core engineering experience and the cost realities coming out of the growing number of commercial nuclear power plants going on the line are supporting this opinion.

Predictions of electrical energy requirements in the United States show an increase of from 200 to 300% in the next 50 years. By that time it is reasonably estimated that perhaps 50% of this requirement will be filled by nuclear generating capacity, provided that nuclear power costs follow the trends now foreseen. Thus, it is not the long-term future of the reactor industry which presents a major problem, but it is the years immediately ahead that are of concern.

The present magnitude of capital expenditures needed for nuclear power plants and a lack of immediate favourable operating costs have resulted in a "wait and see" attitude on the part of many utilities. Some public concern has been voiced over continuing U.S. Government expenditures for atomic power in view of mounting requests for funds in other areas of science and technology.

The real hurdle will come in the next 10 to 15 years. During this time, nuclear plants will be accumulating operating hours and demonstrating their competitive cost positions. The nuclear industry has learnt that it is not challenging a stationary target in its pursuit of conventional power costs. Recent developments in the area of increased operating efficiency and availability of resources have shown that, for the immediate future, generating costs from conventional fuels will not automatically rise as was first thought.

As the practical problems involved in the utility operation of nuclear reactors are becoming evident, there has developed an increasing scepticism as to the ease with which idealistic reactor concepts can be made into useful power producers. As a result, divergent viewpoints are developing concerning the relative emphasis being placed in the U.S. national programme on reactor designs that are obviously of immediate practical nature and on reactor designs of theoretically better nuclear performance, but requiring decades of basic development. As examples of such diversity of thinking are the publicized discussions of the homogeneous versus the heterogeneous reactors, and of converters versus breeders.

Grant Financing of Construction

In this difficult period, the operating utilities, the Federal Government and the nuclear equipment industry have important sustaining rôles to play. It is increasingly evident that substantial federal support for reactor development will continue to be necessary for a number of years. Detailed 10-year programme plans are now being formulated by the U.S. AEC and the reactor industry. The implementation of these programmes should do much to ensure that necessary financial support produces the greatest possible technical advancement.

The large initial capital investment, mentioned earlier, still represents a major impediment to increased participation by the utilities in the nuclear programme at this time. A programme of capital cost assistance for construction plants through government grants to utility operators still appears to be needed. Under such a system, the Government would make a grant proportional to the capital cost difference between a nuclear and conventional plant.

The logic of Government grants for construction of nuclear plants is simple. There appears to be general agreement that a sensible nuclear power programme will involve construction costs of the order of $1,000 million during the next 10 years. Under the grant system about half of this cost could come from utility systems, and the remainder from the federal budget. In effect, the Government will achieve its end purpose—the development of technical and economic data, and the stimulation of an industry—at substantially less cost. A number of varying plans and proposals for such grants have been made, but, as might be expected, there are valid criticisms of these suggestions and, as yet, no action has been taken.

One objection to a grant programme has been the suggestion that it would be necessary for the Government to take title to portions of plants constructed with federal funds. It is not evident why this should be so. As mentioned earlier, the Government would be receiving technical and economic information at approximately half the investment it would otherwise make. The public and private utilities participating in such a programme would not be receiving a windfall, as the plants constructed would not produce power cheaper than current conventional plants. This being the case, it would serve no public need to have the Federal Government maintain strings which, quite properly, would be unacceptable to the largest segment of the power industry. Federal monies have been made available to benefit particular industries in the past and no precedent was set in those cases for taking a lien or retaining title. Examples are the Government support programmes for such activities as airlines, agriculture, and railroads.

As an alternative to a government grant programme, it has been suggested that the utility systems contribute jointly to a development fund which would support the construction of large nuclear plants in the same manner. In the past several years, such action has been taken by groups of private utilities for specific nuclear experimental plants. However, the absence of a regular sustaining activity of this sort makes it difficult for the national programme to include this form of support in its planning. The result has been a spasmodic programme of nuclear plant construction, continuously influenced by the conflicting pressures of public versus private power advocates and national versus private expenditure enthusiasts.

Rôle of the National Laboratories

Recently a discussion has arisen in the United States concerning the future rôles of the industrial and national laboratories in the atomic energy programme. The possible decline of weapons development and budgetary adjustments in Government-sponsored research and development have led to this concern over the respective activities of these laboratories.

The vast Government research centres, which were brought into being with the development of the atomic bomb, have made a monumental contribution to the advancement of peacetime nuclear science in the years that followed. During this same period, private industry in the United States built up substantial facilities, specialized technical sources, and experienced management devoted to this technology.

In many respects, it is incongruous that any of our Government or private laboratories, with outstanding personnel and equipment, should feel threatened by future programme changes in view of the overall desire of the nation to increase its technological capabilities. Obviously, if the national technical effort is going to increase, all available facilities and manpower will be called upon to produce. The fact that a potential problem exists at all is probably an indication of an uncertainty in future planning for the application of the technical strength of the country.

The national laboratories represent a valuable asset which should continue to play a significant rôle in atomic energy. These laboratories should concentrate on basic science and pioneering technology and, where appropriate, conduct reactor experiments utilizing these advantages. The Government laboratories, as in the past, would thus be the source of the major basic scientific and technical advances in atomic energy.

It is interesting to note in this respect that in the past the national laboratories themselves have generally desired to expand their pioneering types of research. Unfortunately, they have been unable to obtain full financial support for such basic programmes, although there has been support for the development of operational hardware.

In parallel, the industrial laboratories of the United States should then carry the burden of introducing the new technologies to practical application. Reactor and engineering development phases would thus be carried out by the organizations having the ultimate responsibilities of manufacturing and selling atomic power equipment.

There are, of course, grey areas of overlap in these rôles. The dividing line between the areas of responsibility is not a fine one. If Government policy considers industry as an essential partner in its reactor programme rather than as a competitor to the national laboratories, a balanced solution in the national interest can be found. Intelligent, long-range planning in both technical and budgetary fields can ensure that both of these partners will contribute the maximum to the development of atomic power in the United States.

Export Markets

by **G. J. STATHAKIS, Manager and K. F. PAULOVICH, Specialist**

(Overseas Power Reactor Applications, Atomic Power Equipment Department, General Electric Company.)

FOR American reactor suppliers, the same considerations apply to overseas markets as to the U.S. domestic market; namely, those areas utilizing large size generating plants with high fuel costs, in general, present the most promising opportunities for nuclear power plants. This immediately leads to the conclusion that most of Europe and Japan, with their relatively high fuel costs and their relatively large electrical growth patterns, are the most attractive.

The European situation has, of course, been somewhat altered in recent months by the surplus coal, and the discovery of oil and gas in the Sahara and France. However, it is generally accepted that the surplus coal situation is somewhat transient, and does not have any great bearing on the nuclear power industry, which has not and is not likely to reach full bloom for one or two decades. The availability of oil and gas from the Sahara and other sources will certainly have some effect on the average fuel costs in Europe, but the basic economic factors underlying the high fuel cost situation in Europe will not be affected appreciably by these discoveries. So long as Europe remains dependent on the importation of oil from countries outside Europe, it is very likely that the fossil fuel costs will remain relatively high. The tremendous rate of growth of Japanese industry and its consumption of electricity also makes Japan a very important overseas area.

In addition to the primary nuclear markets which exist in Europe and Japan, other individual countries offer some potential for nuclear power plants. Accelerated industrialization in developing countries, coupled with high transportation costs and/or high conventional power costs in isolated areas, may justify nuclear power plants; countries such as Brazil and India fall into this category.

The most important consideration, of course, in looking at the long-range future in the overseas areas, is the rôle of American manufacturers relative to that of local industry. It has been considered by many experts that the American manufacturers would play an important rôle during the initial years of the nuclear industry, but would rapidly withdraw from the overseas scene as the local manufacturers developed technology and manufacturing competence in the nuclear field.

The part to be played by American manufacturers, however, may be much greater than anticipated if the total magnitude of nuclear business grows at a reasonable rate. There will certainly be close working relationships between American companies and their overseas licensees during the initial period and for many years thereafter, and it is conceivable that American manufacturers will continue to be directly concerned with overseas nuclear business for an indefinite period of time. We would view the situation as a matter of American manufacturers working closely with their European associates in order to bring to the overseas areas their latest technologies in reactor concepts under the most efficient arrangements.

In this context, General Electric has already seen, put into practice the thesis of close co-operation and participation, to the maximum extent, by local industry. For example, in the construction of the Kahl plant in Germany, Allgemeine Elektricitats-Gesellschaft acted as a prime contractor for the complete plant with General Electric supplying the reactor system and nuclear fuel. However, it should be noted that a significant portion of the equipment in the Kahl plant, including the reactor pressure vessel, heat exchangers, piping, etc., was procured from local German industry and, of course, the erection of all the reactor system was performed by German labour. The conventional part of the plant was supplied by AEG in circumstances which were very similar to those employed in the construction of fossil-fuelled plants. Looking at the Garigliano plant, which is being constructed by SENN in Italy, International General Electric Operations, S.A., of Switzerland, is acting as prime contractor, with General Electric supplying the reactor system, fuel and technical services as sub-contractor. The majority of the equipment is being purchased in Italy from local manufacturers and the construction work will be performed by Italian firms. The overall intent of General Electric at the SENN plant is to arrange to the maximum extent possible for Italian participation within the technical limitations.

The eventual pattern for working relations between American and overseas companies is difficult to predict at this time. Many factors can enter the picture to cause changes in present

business philosophy, such as the amount of basic research and development performed by overseas firms, the currency exchange problems, local regulations regarding permits, licences, import and customs duties, etc. However, it is clear that the overseas market presents opportunities for nuclear power which must be pursued by American industry if the nuclear business is to proceed at the fastest rate of development that is achievable.

One of the major obstacles facing reactor manufacturers competing in the overseas market for nuclear power plants is the lack of adequate third-party nuclear liability protection. The recent passage of nuclear liability legislation in individual countries, such as Germany and Sweden, indicates major attention is being given to solution of this problem in some areas. However, because of the closeness of countries in Europe, standard uniform liability legislation must be available not only for the country in which the reactor is installed, but also in contiguous countries. The passage of the OEEC Convention on nuclear liability alleviates this problem to some extent, but it is only a first step in establishing an adequate system of indemnity in Europe. The next step required is the increasing of the limit-of-liability level and, concurrently, the indemnity level. Until legislation protecting the public is drawn up and private utilities and reactor manufacturers have adequate third-party nuclear liability insurance and/or government indemnity, the liability question will remain an obstacle in developing the overseas nuclear power market.

Many power reactor concepts are currently under development in the United States through the co-operative efforts of the U.S. Atomic Energy Commission, electric utilities and reactor manufacturers. The progress and advancements made in these U.S. reactor development programmes are made available by reactor manufacturers to overseas customers as soon as they are established.

U.K.—U.S. Collaboration

by SIR ALAN HITCHMAN, K.C.B.

(AEA Member for External Relations and Commercial Policy)

AS the Chairman of the United Kingdom Atomic Energy Authority, Sir Roger Makins, said recently, the co-operation now existing between Britain and the United States of America on atomic energy matters is of a cordiality and completeness that can only be compared with that which obtained in the days of our wartime alliance. It is co-operation based on the respect of each country for the achievements of the other and the more firmly based perhaps because, for a time, each of us went a separate road. When the roads rejoined each had much of value to offer the other.

It will be recalled that the wartime partnership of the two countries was severely limited by the McMahon Act, passed by the United States Congress in 1946, which set back co-operation including the flow of information between the U.S.A. and other countries. Much as that barrier between us was regretted on this side of the Atlantic, a highly industrialized country like ours—with large and widespread military commitments and rapidly increasing power requirements—could not afford to ignore this new source of energy. We had to "go it alone."

We did so with such good effect that we were a fully fledged atomic power, in both a military and a civil context, by the time the U.S. Atomic Energy Act of 1954 ushered in a new era of Anglo-American co-operation. We were one of the first countries to benefit from the relaxation of the restrictions on the exchange of information under the new Act and in June, 1955, agreements were signed between our two countries, covering co-operation in the civil and military fields.

By this date, work was already well advanced on Britain's first nuclear power station at Calder Hall and when, later in the same year, the first international conference on the peaceful uses of atomic energy was held under United Nations auspices at Geneva, both our countries were able to display considerable achievements.

There is a great difference of scale in the atomic energy effort of the two countries. In round figures the United States have 10 times as many men and spend 10 times as much money on this work as we do. Despite this disparity I am confident that our friends in the United States do not feel that the traffic in ideas and information is a one-way affair. This sense of complementary effort is undoubtedly helped by the fact that, for good domestic reasons in each case, Britain and the U.S.A. have tended to concentrate initially on two different main lines of development where power reactors are concerned —in our case, the gas-cooled graphite-moderated reactor; in the case of the U.S.A., the light water reactors.

We now have a situation developing in which these national differences of approach are becoming less sharp and clearly defined. The U.S.A. are now keenly interested in the potential of gas-cooled reactors and their programme includes two advanced gas-cooled reactor projects. We, on our side, while maintaining our faith in the gas-cooled reactor for land-based power have, with considerable American help, gone into the light-water field for nuclear submarines. The first propulsion unit is being purchased from the U.S.A., but the Admiralty is placing an order in this country for the second unit.

Specific Fields of Collaboration. Recapitulating some of the main areas of co-operation between our two countries:—

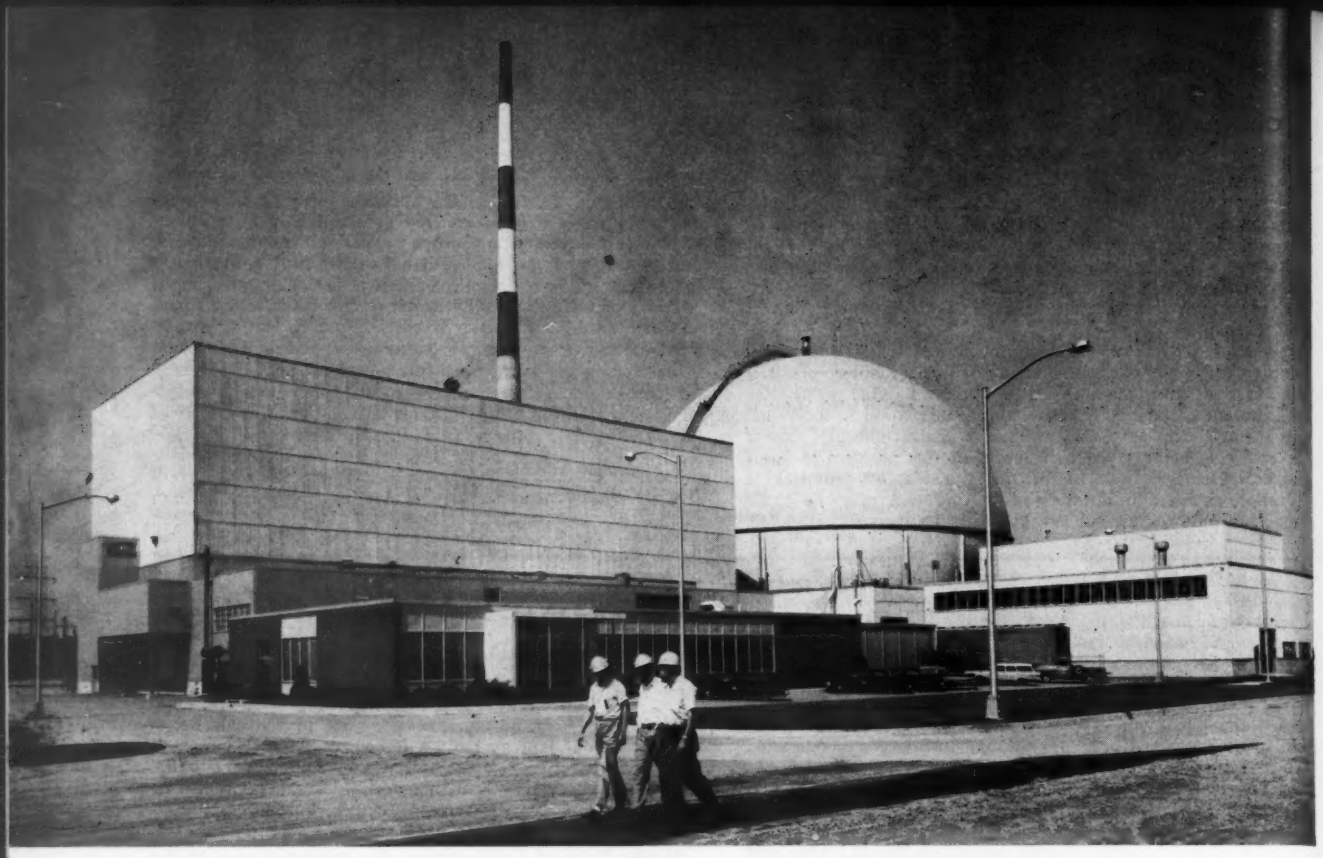
Under our various defence agreements, collaboration on the military side is extremely close, including exchanges not only of information but also of key materials.

Under the civil agreement there are specific arrangements with the U.S. AEC for the exchange of information on the Calder Hall and Chapelcross reactors; on existing fast reactor experiments in both countries; on the advanced gas-cooled reactor at Windscale and the experimental gas-cooled reactor at Oak Ridge, Tennessee. The Authority have a separate agreement with the Power Reactor Development Company (in which a number of leading U.S. firms are associated) for the exchange of information on the respective fast reactors at Dounreay and at Monroe, Michigan. The United States Atomic Energy Commission has reached an agreement with the OEEC Dragon Project (in which Britain is a major partner) for exchange of information about the Dragon high-temperature gas-cooled reactor to be built at Winfrith and the similar type of reactor to be constructed at Peach Bottom, Pennsylvania. In addition, there is a full exchange of information between the Authority and the Commission on controlled thermonuclear reactions and on basic work in all the relevant fields of study.

Collaboration between the Authority and the Commission can and does take many forms. There is a continual traffic of scientists and engineers across the Atlantic in both directions, for short visits, for conferences and discussions or for attachments that can be of months' or years' duration. The number of reports exchanged is in the thousands. There is no doubt that both sides find this sharing and exchange of experience of inestimable value. Neither side suffers under the misapprehension that it it has a monopoly of the good ideas.

Supplementing this relationship, there is a steady growth in the links between private industry in the two countries. Where information is privately owned, both countries recognize the advantages of exchanges being affected on a normal commercial basis. Readers of this journal will be aware of the various technical co-operation and licensing agreements which have been concluded between U.K. and U.S. companies.

It can readily be seen that co-operation between our two countries has extended to a range and complexity that it would have been difficult to foresee and optimistic to prophesy five years ago. It is bringing such tangible benefits that it seems certain to grow and diversify at an ever increasing rate in the years ahead.



Dresden Nuclear Power Station

Outline Design

DRESDEN, the largest all-nuclear plant in the United States—and entirely financed by private enterprise—was designed and constructed by the General Electric Company, who received a complete turnkey contract at a lump sum price of $45,000,000 (without escalation) not including site costs. Associated with them was the Bechtel Corporation, as engineer-constructor, providing design services for the conventional portions of the plant, and all constructional services. G.E. also had the responsibility of fuel loading and start-up of the reactor, the operation during the "working-up" period, and the complete training programme for operating personnel.

Owners of the station are the Commonwealth Edison Company, with whom are associated the Nuclear Power Group, Inc., the composition of which is given in Table 1.

Table 1. Members of Nuclear Power Group, Inc.

American Gas & Electric	Pacific Gas & Electric
Central Illinois Light	Union Electric Company of Missouri
Commonwealth Edison	Bechtel Corporation
Illinois Power	
Kansas City Power & Light	

Dresden is located some 47 miles S.W. of Chicago, at the junction of the Des Plaines and Kanakee rivers, the nearest concentration of population being the township of Morris, eight miles west. Construction began on March 1, 1957, and was completed within two weeks of schedule—and well ahead of its contract date—within 32 months. Fuel loading began on October 13, 1959, and criticality was achieved in the early hours of October 15.

Dresden is a dual-cycle boiling water reactor, with an output of 625.9 MWt. The net electrical output of the station is 180 MW, at 0.85 p.f., with cooling water at 75°F. Main design parameters are contained in the tables on the pull-out drawing opposite page 437.

Core

The core is an approximate cylinder 10.7 ft equivalent diameter, and 11.2 ft. high, consisting of 452 fuel elements arranged in a square lattice. The bottom core support plate rests on a ring girder supported by columns attached to the bottom of the reactor vessel, which also supports a cylindrical core guide support (also acting as a thermal shield) which locates the upper grid structure locating the fuel elements and control rods.

Control

There are 80 control rods, of cruciform section boron stainless steel (0.2% boron), 6.5 in. square, and located at 9.96 in. centres, within a circle 8 ft 3 in. diameter in the central region of the core. Operation is from beneath the core, the rods being raised to reduce reactivity, the total travel being 8 ft 10 in. Hydraulic operation is used, with a positive latching mechanism to eliminate the drift commonly associated with hydraulic mechanisms, and the necessity for high-pressure running seals and the disadvantage of "piston" effect are eliminated by enclosing each mechanism in a thimble extension to the pressure vessel. The scram time is 2.5 sec.

In addition, there is a back-up system of chemical poisoning for injecting a solution of sodium pentaborate which will give a boron concentration of about 690 ppm in the cooling water when uniformly dispersed, reducing reactivity by some 20% $\Delta k/k$.

Fuel Elements

The fuel elements and their manufacture have been fully described on a previous occasion (*Nuclear Engineering*, June, 1959, p. 253) and it will suffice to say that each element consists of uranium oxide pellets clad in 0.03 in. Zircaloy-2, to form rods approximately 0.56 in. diameter. There are 36 rods in each element, in a 6 × 6 array, the whole being enclosed in a Zircaloy-2 shroud of square cross-section, radiused at the corners.

Pressure Vessel

The reactor vessel is of carbon steel, clad with stainless steel, and is 12 ft 2 in. inside diameter, and 40 ft 10 in. high, the plate thickness varying from 5¼ in.-5⅝ in. There are four 22 in. inlet nozzles near the bottom of the vessel, and 12 16 in. outlet nozzles above the core, and just below the turning vane, or flow baffle which is fixed above the core. The design pressure for the vessel is 1,250 p.s.i.

Coolant Circuit

As will be seen from the flow diagram, a dual cycle is used for power production. Primary coolant enters the reactor at a rate of 25.7×10^6 lb/h, the inlet temperature being 505°F, leaving as a steam-water mixture at 547°F for the main steam-and-water drum, where separation takes place, and the steam passes to the turbine at a rate of 1,405,000 lb/h. The water passes through the coolant circulating pumps and is returned to the reactor inlet via secondary steam generators, in which it is cooled down to the reactor inlet temperature. The heat given up in this process is used to generate another 1,191,200 lb/h of secondary steam at about one-half the pressure of the primary steam, entering the turbine at a later stage.

There are four main coolant pumps of the canned-rotor type, with 600 h.p. motors. Each pump is rated at 16,500 gal/min against a developed head of 103 ft.

The main steam drum is 7 ft 11 in. i.d., and 67 ft long. Centrifugal separators and chevron driers ensure that moisture content of the steam does not exceed 0.1%.

The four secondary steam generators are of the vertical-shell type, with the primary water in an inverted U-tube bundle; each is rated at 297,800 lb/h at 510 p.s.i.a., both shell and tube sides being designed for a pressure of 1,250 p.s.i.

Turbine and Condenser

The turbine is a three-cylinder tandem compound machine, running at 1,800 r.p.m. rated at 192 MW with 27½ in. vacuum. The L.P. cylinder is double-flow, and exhausts directly into the main condenser, which has a surface area of 120,000 ft². If necessary, steam can be dumped (at a rate of 125% of normal flow) direct to the condenser, via a system of reducing valves and desuperheating sprayers.

If, for any reason, the main condenser is not available as a heat sink, there is an emergency condenser located above the main steam drum, in which primary heat is transferred to water which is evaporated to atmosphere. The capacity is 6% of the normal reactor output so that it will deal with decay heat, in emergency, sufficient water being stored for an eight-hour period without attention, or longer if the water supply is replenished.

Containment and Shielding

Reactor, steam drum, secondary steam generators, emergency condenser and recirculating pumps are all built into a concrete structure which provides the necessary shielding. This structure is enclosed within the containment, which consists of a steel sphere 190 ft diameter, varying in thickness from 1.25-1.4 in., and weighing some 3,500 tons.

The sphere was designed to withstand an internal pressure of 29.5 p.s.i., and tested to 37 p.s.i., with negligible leakage. Air locks for equipment (8 ft × 8 ft), personnel (6 ft × 2 ft 6 in.) and an emergency escape lock 2 ft 6 in. diameter, are provided. During shut-down a bolted opening 16 ft diameter can be used for handling heavy equipment.

Throughout the design of the station, the design target for shielding was a maximum exposure of 100 mrem/week, taking into account all types of radiation—neutrons from the reactor, gammas from fuel, and radiation from N^{16} created by neutron bombardment of oxygen and nitrogen in the cooling water. The design is based on a maximum dose rate which varies with the anticipated exposure time, varying from 0.5 mrem/h in uncontrolled access areas to 12 mrem/h in areas where access is infrequent, such as secondary steam generator equipment cells when shut down for maintenance under operating conditions.

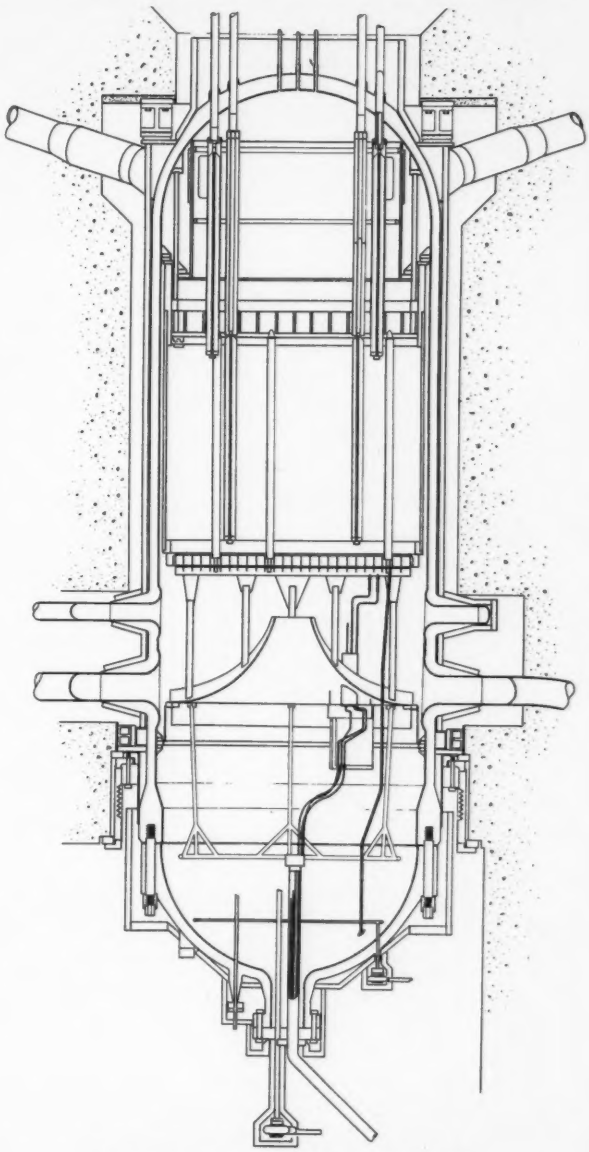
Site Layout

Next to the sphere, which dominates the site, the largest unit is the turbine building (200 ft × 90 ft × 90 ft) which is located to the west of the sphere and contains, in addition to the turbine and condenser, auxiliaries such as feed pumps and heaters, condensate demineralizers, etc. The main control room is also housed in this building.

Details of the other buildings on the site are shown on the site plan, and the pull-out drawing opposite page 436.

Fuel Handling

After flooding the refuelling canal above the reactor and removing the vessel head, the spent fuel is removed by long-handled grappling tools, assisted by crane, the depth of water above the core being nearly 40 ft. The fuel is then transferred



Cross-section through Dresden pressure vessel.

by a rack carrier which traverses the canal, and is then lowered by a hoist in a basket through a vertical discharge tube through the containment to a point well below ground level; a vertical lock arrangement ensures that the system is kept full of water at all times. The fuel basket now travels on horizontal rails through a water-filled tunnel to a point underneath the fuel-handling building, where it is removed to a storage pool until it is cool enough to be shipped to the processing plant. Storage space is provided for 120% of the reactor loading.

Fresh fuel from the fuel store (which has storage facilities for a complete core loading) follows the same path in a reverse direction.

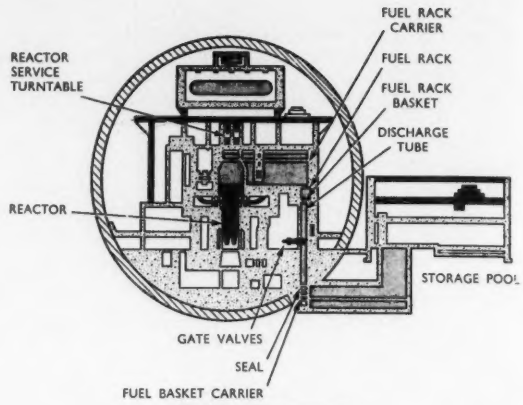
Waste Disposal

Solid wastes, such as spent resins from demineralizers, defective equipment beyond repair, and miscellaneous contaminated rubbish, will be compacted, boxed, and buried in underground storage vaults.

Liquid wastes are treated by various methods, according to the life of the activity and its concentration. Small volume, high-activity wastes are stored indefinitely; low-level large-volume wastes are sluiced to the river after monitoring and analysis.

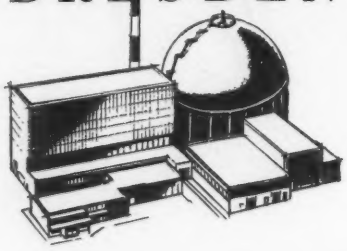
Ventilation air is monitored, filtered where necessary, and discharged to the stack, except in the case of maintenance and service buildings where roof-level discharge is permissible. The most interesting problem with a boiling water reactor is, of course, that of the off-gas from the steam which appears at the air ejector and the gland seal condensers. Since this gaseous activity is normally short-lived, a five-minute hold-up in buried piping is sufficient to reduce concentrations to innocuous levels. These gases are, however, continuously monitored in case of a fuel element failure and, in that event, the hold-up period would allow of plant shut-down and valving-off the gas discharge before release to the stack took place.

The stack discharge as a whole is monitored, and the maximum exposure off-plant is not expected to exceed 10 mrem/year. Since the licensing regulations fix a maximum of 500 mrem/year, this will give a fair margin for off-standard conditions.



Fuel handling system.

DRESDEN



Design Philosophy

by V. D. NIXON

(Manager, Projects Operation Section, Atomic Power Equipment Department, General Electric Company, San Jose, California.)

THE design parameters of the Dresden core can be described through a logical consideration of the heat transfer and fluid flow limitations initially established. These limitations concern:—

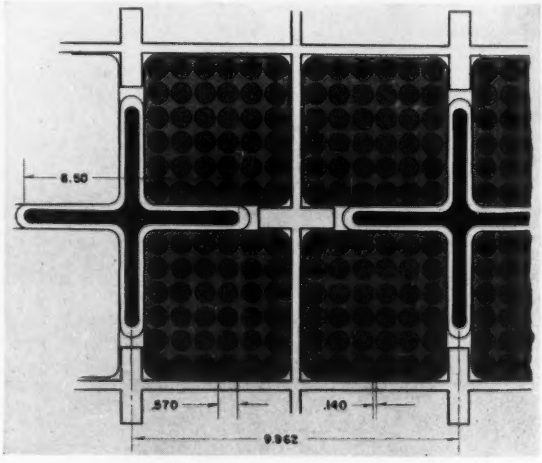
Maximum fuel temperature,
Void or steam content in the core, and
Heat transfer conditions resulting in burn-out.

To ensure against melting of the uranium dioxide fuel during a power excursion, its maximum temperature is limited to less than 5,000°F. Within this limitation, the maximum heat flux at the fuel rod surface is 350,000 B.t.u./ft², h. The corresponding maximum fuel temperature is about 4,800°F based on a thermal conductivity for UO_2 of 1.1 B.t.u./ft², h,°F.

The overall design hot spot or flux peaking factor which expresses the effects of power distribution in numerical form is 3.6, which allows for a steady state power excursion of 25% over rated power. Overpower scram signal setting is 125% of rated power. In addition, allowance is made for:—

1. Axial power distribution.
2. Radial power distribution.
3. Fuel burn-up.
4. Local flux peaking due to water gaps between channels and between fuel rod segments.

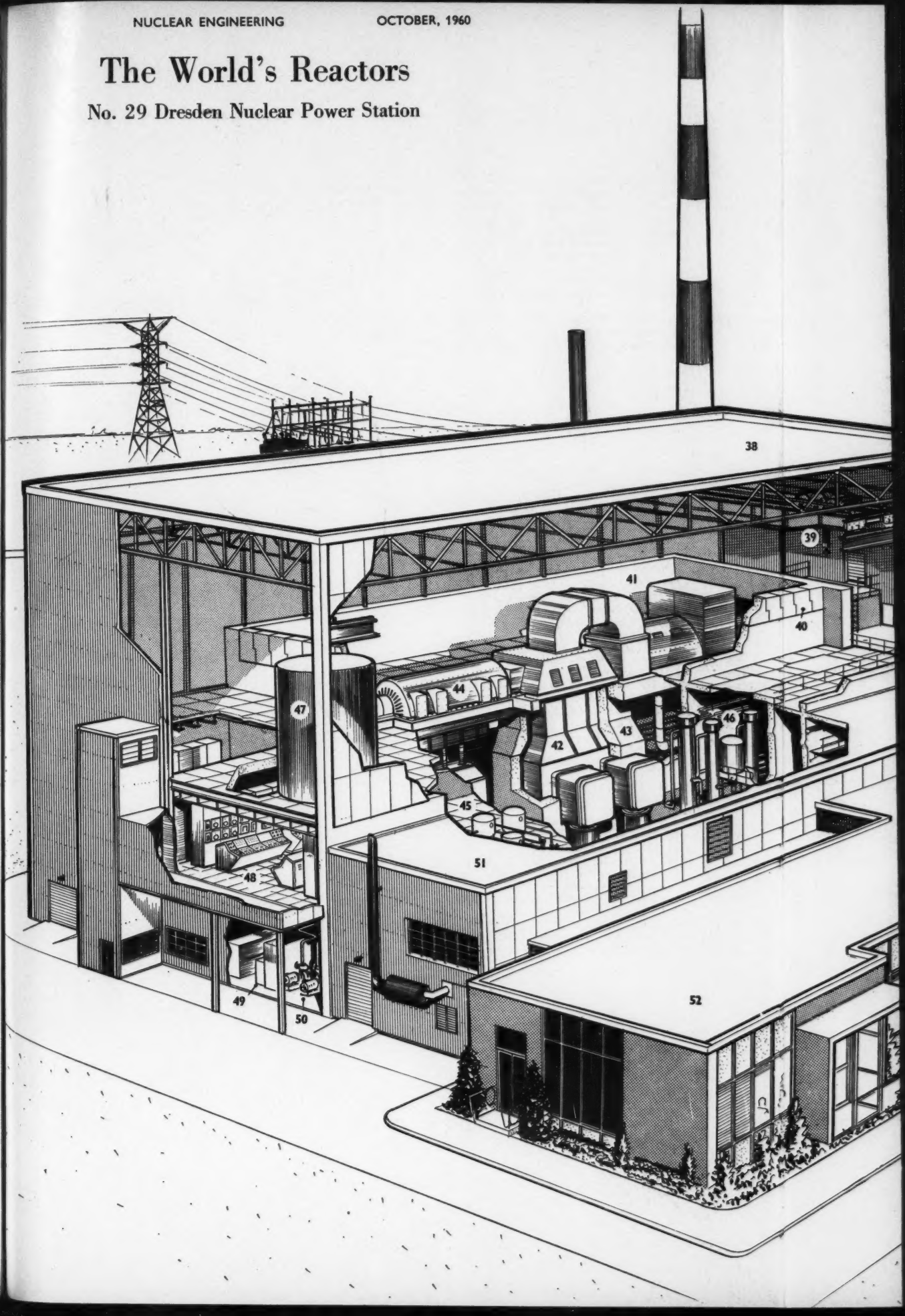
All peaking factors are conservatively considered to occur in coincidence. Because the neutron flux distribution will be subject to controlled variation during the life of the reactor, the hot-spot factor is history-dependent, and the magnitude chosen is such that the actual values during operation will never exceed the design value. Since the maximum heat flux at 25% overpower is 3.6 times the average heat flux at rated power, the average heat flux is calculated to be approximately

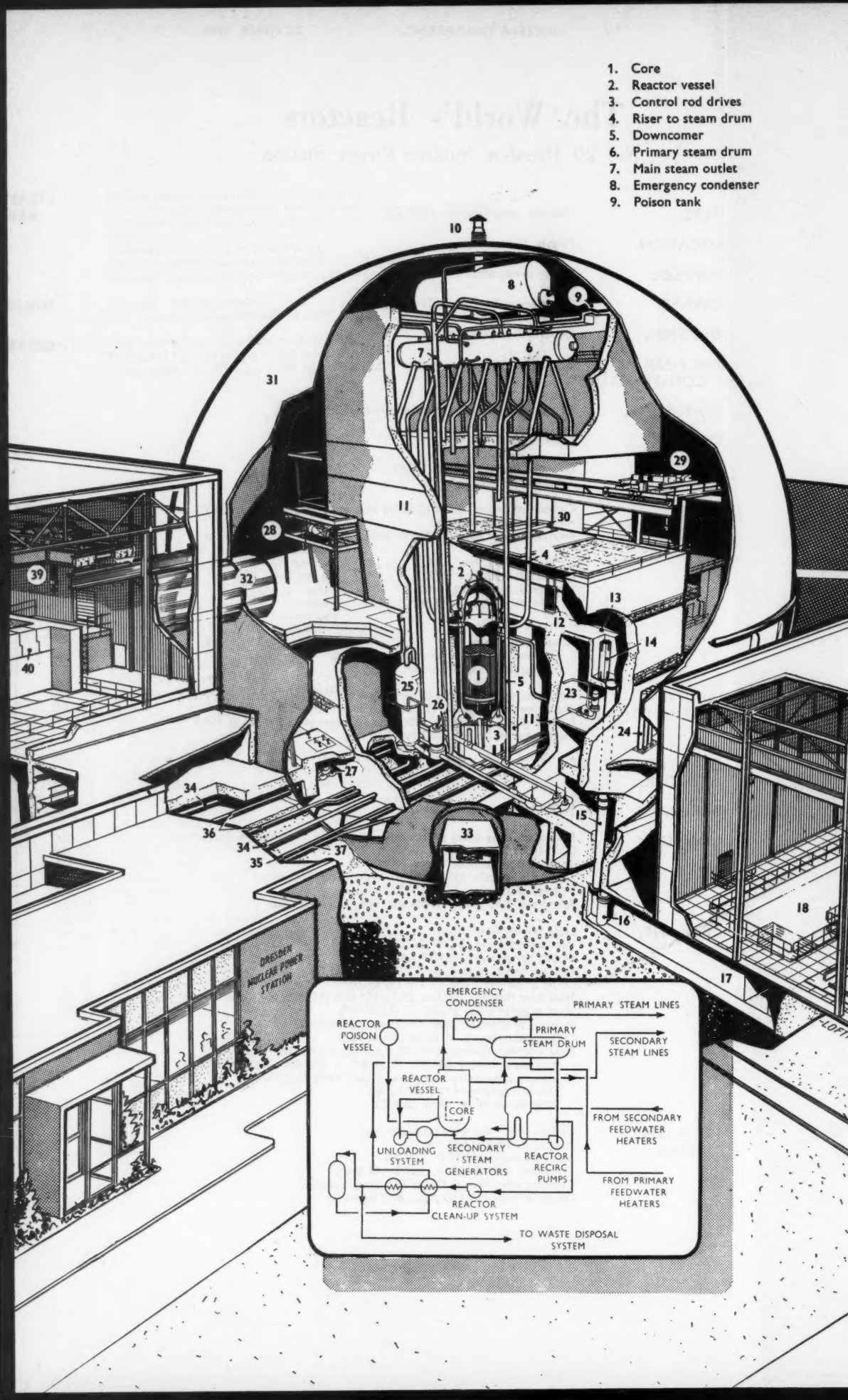


36-rod fuel assembly (dimensions in inches).

The World's Reactors

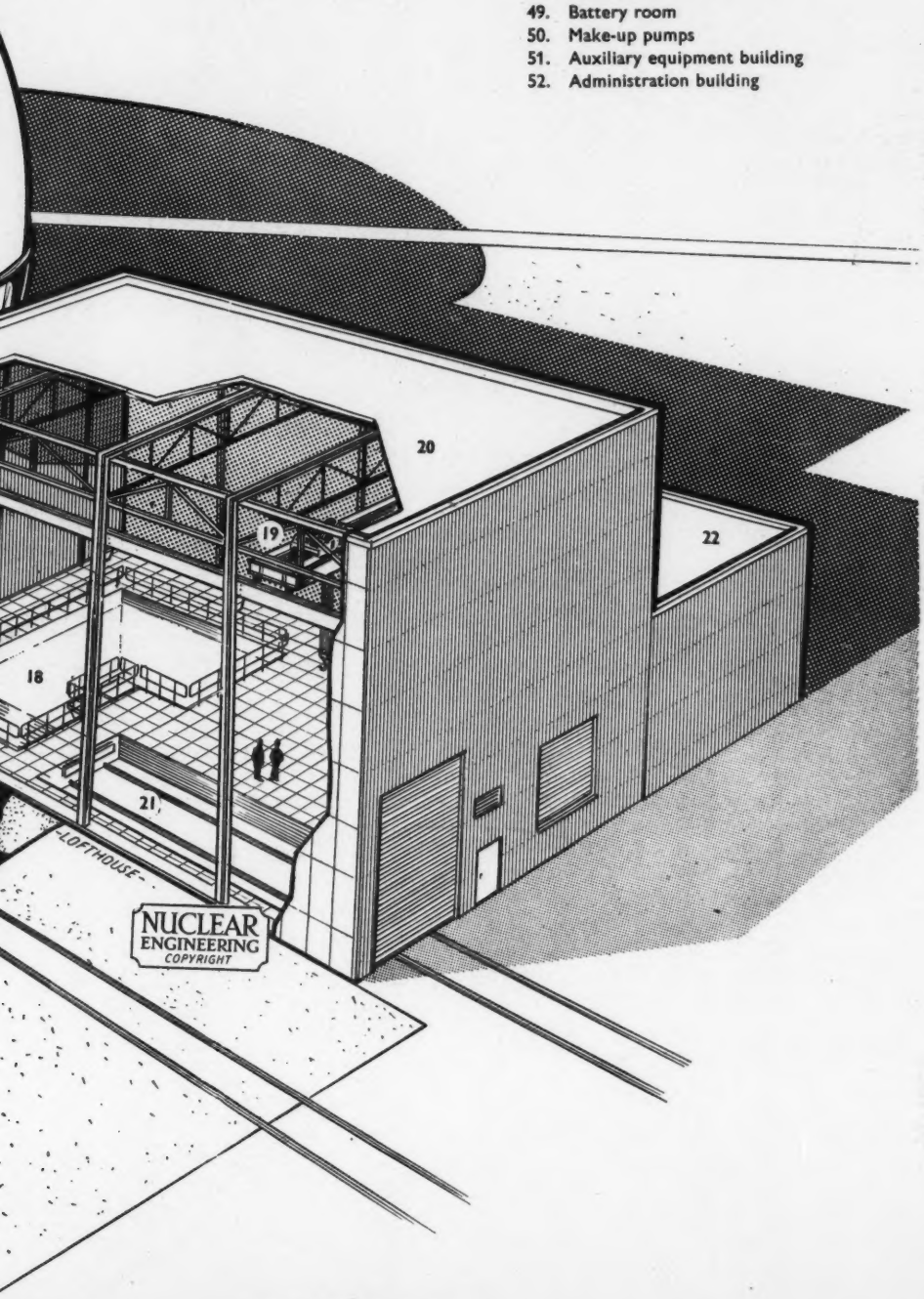
No. 29 Dresden Nuclear Power Station




1. Core
2. Reactor vessel
3. Control rod drives
4. Riser to steam drum
5. Downcomer
6. Primary steam drum
7. Main steam outlet
8. Emergency condenser
9. Poison tank
DRESDEN NUCLEAR POWER STATION
EMERGENCY CONDENSER
PRIMARY STEAM LINES
REACTOR POISON VESSEL
PRIMARY STEAM DRUM
SECONDARY STEAM LINES
REACTOR VESSEL
CORE
FROM SECONDARY FEEDWATER HEATERS
UNLOADING SYSTEM
SECONDARY STEAM GENERATORS
REACTOR RECIRC PUMPS
FROM PRIMARY FEEDWATER HEATERS
REACTOR CLEAN-UP SYSTEM
TO WASTE DISPOSAL SYSTEM

KEY

10. Vent to atmosphere
11. Concrete biological shield
12. Fuel rack carrier
13. Fuel transfer canal
14. Fuel rack basket
15. Fuel transfer tube
16. Fuel transfer carrier
17. Fuel transfer tunnel
18. Fuel cooling pool
19. Fuel building service crane
20. Fuel handling building
21. Transit bay
22. New fuel storage vault
23. Unloading pump
24. Cooling water heat exchanger
25. Secondary steam generator
26. Recirculating pump
27. Isolation valves
28. Moveable hoist
29. Reactor service crane
30. Reactor service gantry
31. Containment vessel
32. Equipment access
33. Personnel access
34. Primary steam to turbine
35. Primary feed water return
36. Secondary steam turbine
37. Secondary feed water return
38. Turbine hall
39. Turbine hall service crane
40. Removeable concrete shielding
41. Turbines
42. Condenser
43. Condenser shielding
44. Generator
45. Demineralizers
46. Feed water heaters
47. Condensate storage tank
48. Control room
49. Battery room
50. Make-up pumps
51. Auxiliary equipment building
52. Administration building



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The World's Reactors

No. 29 Dresden Nuclear Power Station

TYPE: Thermal, heterogeneous (BWR).

LOCATION: Morris, Illinois.

PURPOSE: Power production.

OWNER: Commonwealth Edison Company.

DESIGNER: General Electric Company.

ENGINEER-CONSTRUCTOR: Bechtel Corporation.

STATUS: Critical October 15, 1959; full power June 29, 1960.

OUTPUT: 625.9 MWt, 180 MWe.

FUEL: Enriched uranium oxide (1.5%).
Total loading 760 kg U^{235}.

CORE: Cylindrical prism 10.7 ft (3.26 m) diameter, 11.2 ft (3.41 m) high, with 452 fuel assemblies.
Average power density: 29 kW/litre.

ε =cold: 1.03; hot: 1.04.
p =cold: 0.85; hot: 0.77.
k_∞ =cold: 1.135; hot: 1.15.
τ cold: 40 cm²; hot: 70 cm².

Excess reactivity for:

Burn-up	7.5%.
Xe and Sm	4%.
Temp.	−0.5%.
Voids	1%.
Shutdown margin	3%.
Total	15% $\Delta k/k$.

Neutron flux:
Thermal, average: 3.15×10^{13} n/cm², sec.
Fast, typical: 1×10^{14} n/cm², sec.

FUEL ELEMENTS: Rod bundle; 36 rods (6×6) per element, at 0.71 in. (18 mm) centres. Oxide pellets 0.494 in. (12.54 mm) diameter, clad in Zircaloy-2, 0.03 in. (0.076 mm) thick. Outside diameter of rod: 0.563 in. (14.3 mm). Active length 105.2 in. (2.67 m).

CONVERSION FACTOR: 0.6.

BURN-UP: 10,000 MWd/t (equilibrium core).

CONTROL: Boron stainless steel cruciform rods, 6.5 in. (165 mm) square × 0.375 in. (9.52 mm) thick.
Total number of rods: 80.
Total rod worth: 15% $\Delta k/k$.
Withdrawal rate: 6 in./sec (152 mm/sec).
Scram time: 10% travel: 0.6 sec; 90% travel: 2.5 sec.

COOLANT: Light water, dual cycle.
Inlet temperature: 505°F (262.75°C).
Outlet temperature: 547°F (286°C).
Reactor pressure (average): 1,010 p.s.i. (71.0 kg/cm²).
Drum pressure: 1,010 p.s.i. (71.0 kg/cm²).
Mass flow through reactor: 25.7×10^6 lb/h (11.66×10^6 kg/h).
Heat transfer area: 21,500 ft² (2,000 m²).
Velocity in channel: inlet 4.5 ft/sec (1.35 m/sec).
Outlet (wet steam): 11 ft/sec (3.4 m/sec).
Heat transfer: (average) 99,200 B.t.u./ft², h (269,000 *k* cal/m², h), (maximum rated) 280,000 B.t.u./ft², h (759,000 *k* cal/m², h).
Maximum element temperatures: Fuel: 4,000°F (2,205°C).
Internal cladding: 675°F (357.2°C).
External cladding: 580°F (304°C).

PRESSURE VESSEL: Carbon steel clad with stainless steel.
Height: 40.88 ft (12.46 m).
Inside diameter: 12.17 ft (3.71 m).
Thickness: 5.25–5.625 in. (133.3–143 mm).
Cladding thickness: 0.375 in. (9.5 mm) nominal.
Design pressure: 1,250 p.s.i. (87.9 kg/cm²).

STEAM RAISING:	Primary steam flow to turbine: 1.4 × 10^6 lb/h. Secondary steam generated in four heat exchangers with inverted U-tube primary circuit and shell secondaries. Height: 27 ft 9 in. (8.46 m). Diameter (internal): 5 ft 3 in. (1.6 m). Secondary steam generated: 1.19 × 10^6 lb/h (541 te/h). Secondary steam pressure: 475 p.s.i. (33.4 kg/cm²).
SHIELDING:	2 in. steel thermal shield. 11 ft concrete biological shield at sides.
CONTAINMENT:	Steel sphere. Diameter: 190 ft (57.912 m). Thickness: 1.25–1.4 in. (31.7–35.5 mm). Design pressure: 29.5 p.s.i. (2.075 kg/cm²) at 250° F (121.1°C).

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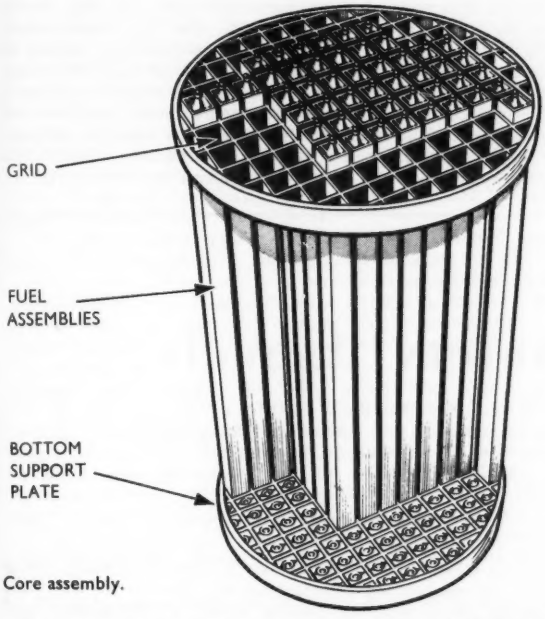
95,000 B.t.u./ft², h. The average power density, based on the total core volume, is 29 kW/litre. This corresponds to 4 kW/ft of fuel rod length.

The void, or steam content, in the core is established as the result of a compromise between physics limitations, heat transfer limitations, and economics. If the void content is increased, enrichment, control requirements, and the void coefficient of reactivity increase. On the other hand, flow rate, the ratio of secondary to primary steam flow, heat exchanger requirements, cycle heat rate, and allowable heat flux decrease. Studies balancing these effects resulted in the selection of a primary to secondary steam ratio of 1.2 : 1 and a total coolant flow rate through the core of 25.7 million pounds per hour.

Corresponding average coolant conditions in the core are an inlet temperature of 505°F (equivalent to a sub-cooling with respect to saturation at 1,000 p.s.i.a. of 51 B.t.u./lb), and an inlet velocity of 4.50 ft/sec. The velocity increases to 4.75 ft/sec, as the water expands to saturation conditions, about 5 ft above the core inlet. In the top 4 ft of the core steam is generated,

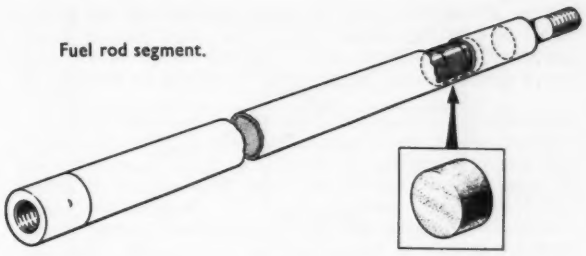


Core assembly.

resulting in an exit quality of about 5% by weight and an exit void of about 50%. The average exit mixture velocity is about 9 ft/sec. Orifices at the lower end of each channel distribute the coolant flow among the channels approximately in proportion to the heat transfer load, to maintain uniform exit quality across the core. The pressure drop across the core is 7 p.s.i.

In every boiling system there is some heat transfer rate which results in steam blanketing of the heat transfer surface. Because steam is a poor conductor of heat relative to water, the surface temperature increases accordingly. This condition usually results in melting of the heated surface (burn-out). The heat flux which results in this appears to be primarily a function of the local quality, decreasing with increasing quality. At the point in the Dresden core where the worst combination of heat flux and quality occur, the heat flux which would cause burn-out is 2.7 times the actual heat flux. This factor represents a large safety factor, hence, from the heat transfer point of view, the Dresden core is said to be centre temperature limited (centre of fuel).

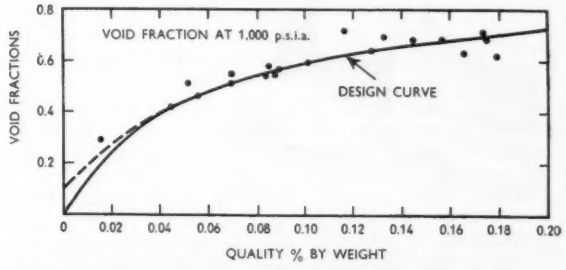
A few years ago there was little experimental information upon which to design a reactor core to meet the centre fuel temperature, void and burn-out limitations. Today the position is much improved, and many major areas of uncertainty with respect to heat transfer and fluid flow have been removed.



Fuel rod segment.

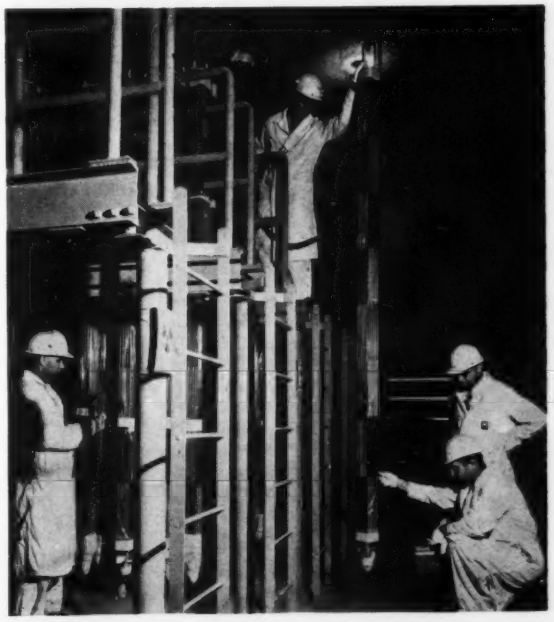
Prototype fuel assemblies containing 9 rods loaded with UO_2 pellets of the same nominal dimensions as used in Dresden were tested in the Vallecitos Boiling Water Reactor. The in-pile fuel tests demonstrated that no damage to the fuel element results from heat fluxes as high as 370,000 B.t.u./ft-h. Since the maximum heat flux in Dresden is 350,000 B.t.u./ft²h, even at 25% overpower, these tests confirm the initial design criteria that no melting will result under the worst operating conditions.

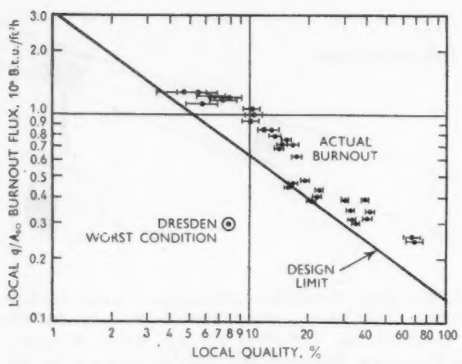
It was mentioned that flow conditions were selected to yield a specific void content and reactivity associated with this void. The void fraction at any point in the core is a function of the local weight fraction of steam as well as the relative velocity or slip between the steam and water. At the University of Minnesota, Argonne National Laboratory and at the General Electric test facility in San Jose, intensive efforts were made to understand the two-phase flow phenomenon—and in particular to measure void fraction as a function of weight fraction of steam. The density of the steam-water mixture is measured by its ability to attenuate gamma rays. Below is shown the design curve that was used to determine void fraction and on the abscissa, weight fraction or steam

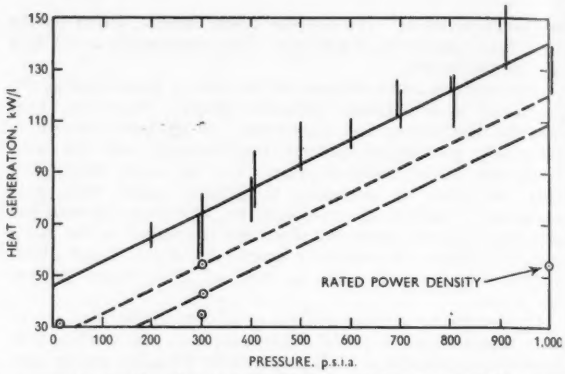


Above.—Void fraction as a function of steam quality.

Below.—Fuel assemblies being inspected.





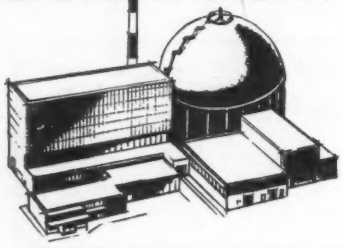


Left.—Burn-out tests on a vertical tube showing the large safety margin present in the Dresden design. Right.—Hydraulic instability tests demonstrating the decrease in sensitivity with pressure to pressure variations. Vertical lines represent regions for which stability was obtained and full line limit of power density tolerable. Dotted lines and point are for Borax III, IV, and EBWR.

quality. The test was conducted with a horizontal pipe operated at 1,000 p.s.i.a. with no heat transfer. The void, at any quality, would tend to be less in a vertical pipe.

In order to investigate those conditions which result in vapour binding and burn-out, an intensive test programme was conducted with the heat transfer facility at San Jose to confirm the design curve used for the Dresden reactor (see above). On the ordinate is plotted the heat flux at burn-out and on the abscissa, the local quality. The heater element used was a vertical thin-walled ¼-in. stainless-steel tube 9 ft long, cooled on the outside by forced circulation. The flow velocity, boiling length, hydraulic diameter and pressure were selected to simulate Dresden conditions. The limits on each data point indicate with 95% confidence the possible error in calculated quality. One point of actual burn-out resulting in destruction of the heater tube is noted. The other points represent indicated burn-out and hence are slightly conservative. These data clearly confirmed the design limit which had been devised.

DRESDEN

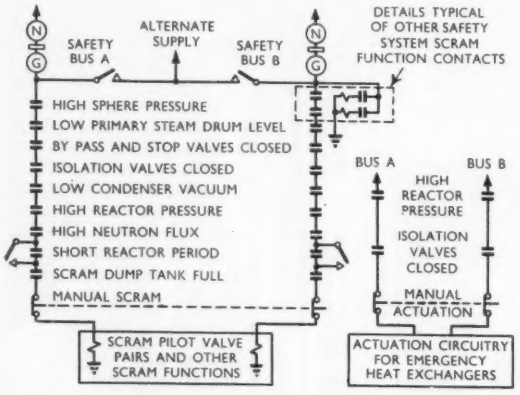


Safety Aspects

By V. D. NIXON

Manager, Projects Operation Section, Atomic Power Equipment Department, General Electric Company, San Jose, California.)

PRINCIPAL objectives in the design of the Dresden safety system were to achieve high reliability for continuity operation and to produce a system shutdown upon approaching unsafe conditions, i.e., eliminating false shutdowns. The concept of a dual safety system was evolved to meet these basic requirements. By this means, two separate power supplies (M.G. sets from two separate auxiliary buses) are fed through several relay contacts to pairs of safety system pilot valves. If power is lost to both members of a pair of valves, a group of three control rods will be inserted rapidly into the reactor. In this way, through many pairs of control valves, all 80 control rods would be rapidly inserted to shut down the reactor if power to these scram holding circuits or pilot valves was removed for any one of several scram conditions.

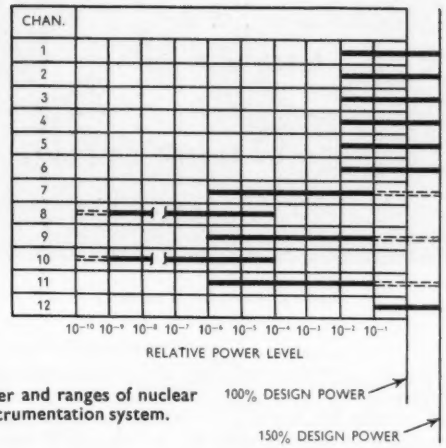


Automatic scram signal system.

In this way, half the dual safety system could probably be de-energized through equipment malfunction without scramming the reactor. If, in the active half of the safety system, either one of a pair of scram function sensors indicated the approach to an unsafe condition, the other half of the system would be de-energized and a complete scram would be obtained. Thus, any one safety system component can fail without scramming the system or compromising its safety.

On the left is illustrated the automatic scram functions the Dresden safety system provides. In addition, it indicates the actuation signals for bringing on the emergency heat exchanger system. Each contact (as shown in the detail area) is actually a relay contact where the relay in turn is de-energized in general by the opening of one scram contact. In this way, for the most part, there are then four independent sensors; two sensors for each scram function in each half of the dual channel system. The detail of the high sphere pressure function is typical. The entire safety system is thus "fail safe," with each sensor annunciated.

When both halves are de-energized, the scram solenoid-operated air pilot valves and air-operated hydraulic scram valves are de-energized, to produce hydraulic scram motion of the drive mechanisms and insertions of the control rods within

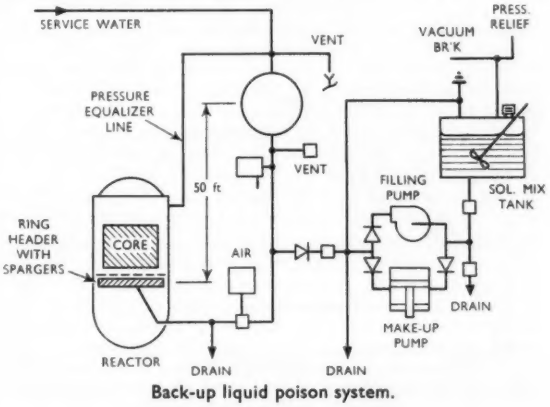


Number and ranges of nuclear instrumentation system.

approximately 2 sec. In addition, all sphere ventilation is closed off to eliminate the possibility of activity leakage should a major accident have occurred. The vent openings are the largest openings in the sphere. At the same time, secondary steam flow is closed off to prevent too rapid a drop in primary system pressure and temperature. The pressure regulator intermittently passes some steam to the condenser to keep the pressure from rising above 1,000 p.s.i. The operator has, as well, the option of producing a scram himself should he decide plant conditions justify such a move.

If a high sphere pressure should arise, it would occur presumably because of a rupture somewhere in the primary system. Because of this, as sphere pressure exceeds atmospheric pressure by approximately 4 in. Hg, a reactor scram would occur. If water level in the primary drum drops too far below normal working level, the system will be scrammed. If the reactor steam supply system is completely bottled up, pressure will continue to rise in the reactor vessel, and because of the pressure effect on subcooling and directly on steam volume, reactor power will tend to increase. To guard against the occurrence of such an unsafe condition, if the primary isolation valves are closed, a reactor scram will occur (as the operator attempts to close these valves). This normally would never be done in practice. One valve at a time may be closed for test purposes, but if both valves are moved off their full-open position by approximately 30% of travel, a scram will result.

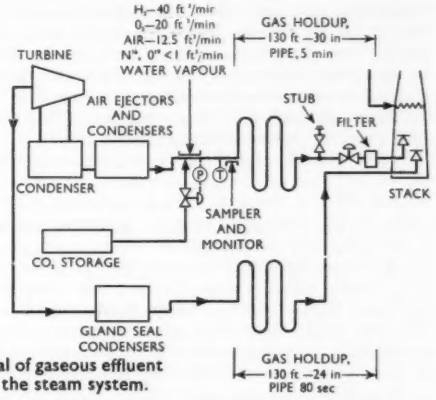
Of perhaps even more concern is the possibility of very rapid closure of the turbine by-pass and stop valves. These valves can be closed in the matter of only a few seconds compared with the 1-2-min closing time of the isolation valves, and might occur from failure of the turbine oil supply. For such a rapid closure, reactor pressure would immediately begin rising at a rate of approximately 18 p.s.i./sec. A scram signal should result immediately from the closure of these valves, rather than



Back-up liquid poison system.

waiting for the high flux signal at 125% rated power, or the reactor pressure scram to come into action at 1,050 p.s.i. The high flux or reactor pressure scram functions will, however, be available to produce a scram should reactor pressure rise continue.

Since the main system heat sink is the condenser, if condenser vacuum is lost it is necessary to shut down the system, and low condenser vacuum protection is provided to scram the system if vacuum falls from the normal working level to approximately 23 in. If vacuum continues to drop to 19 in., turbine primary admission and secondary control valves are closed off. If vacuum is dropped further to 7 in., the turbine by-pass valve is closed as well; thus the condenser is completely closed off at this vacuum level to avoid opening the condenser rupture diaphragm. Of basic concern as a measure of reactor power is the neutron flux signal. If reactor power gets excessive, which could lead to a fuel melting condition or a fuel burnout condition, it would be possible to damage portions of the reactor core. For this reason, if flux exceeds approximately 125% of rated flux a scram is called for. This serves as a back-up to a short reactor scram function



Disposal of gaseous effluent from the steam system.

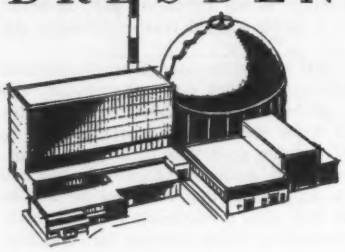
which would normally scram the reactor under this start-up accident condition well before the power range is reached. Such a start-up accident is not visualized as an easily achievable condition, but serves only as a design limit for safety system response criteria. The short reactor period function is by-passed in the power range to eliminate the possibility of nuisance scrams from boiling fluctuations in the reactor.

The final scram function is that which results from filling the scram dump tank. In order to limit system leakage following a scram, and to provide some back pressure to minimize flashing effects in the hydraulic system, a 100 p.s.i. scram dump tank is used with an automatic level control that normally keeps it almost empty. However, should a few scram valves fail, the leakage rate would be such that the dump tank would begin filling fairly rapidly and, should it become completely full, no free volume could exist to accommodate a scram should the necessity arise. For this reason, several alarms are provided as the level rises and when only approximately 1½ scram volumes remain, a scram is initiated.

Activation of the emergency heat exchanger is achieved only by high reactor pressure or by closure of the isolation valves. The use of the emergency heat exchanger, together with the pilot-operated relief valves, is more than sufficient to limit the pressure peak when fast isolation of the system occurs to a point below the lowest safety valve setting. However, these pilot-operated relief valves are on the downstream side of the isolation valves, and upon closure of the isolation valves are no longer available for use. For this reason, a closure of the isolation valves also operates the emergency heat exchangers. It is noted that no scram occurs from failure of power to the recirculating pumps. Should pump power be lost, the plant power reduces automatically as circulation falls.

The reactor can continue to operate at a reduced level of about 50% with natural circulation if operating at rated load prior to power failure.

DRESDEN



Full Power Operation

POWER operation of Dresden commenced on April 15, 1960, and the initial stages concluded on May 1, after approximately ten days steady operation at half power, when the reactor was shut down for a gamma scan of the core, and general preparations for 10 full-power test programme.

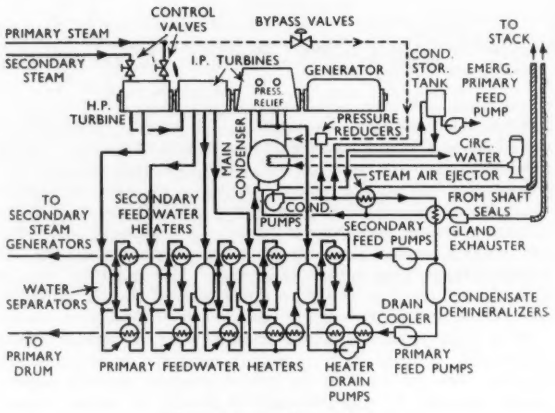
Gamma scanning was carried out with the head removed, and 84 elements were scanned at 2-in. vertical intervals. Prearranged core power distribution symmetry during steady-state operation ensured that these elements were representative of 212 locations. Local power distribution was obtained by scanning individual portions of a disassembled element. From all this data, the corner rod peaking factor was obtained. This was found to be 3.6—somewhat below the design figure.

Preparations for full-power testing included modifications to the control rod guide tube, to reduce the amount of water by-passing the fuel elements. Four additional fuel elements were added to the core, this being the maximum permissible number to remain within the limits of " stuck rod " shutdown. This brought the total number of elements for full-power operation to 452. Other modifications included the installation of some additional local shielding, and the rectification of a slight leak at the head seal, discovered on hydrostatic testing after head replacement, and discovered to be due to scale under the O-ring. A faulty thread in one flange hole was also repaired.

Working-up

The working-up period commenced on June 21, and there were some delays due to the control rod hydraulic system pressure regulators, and miscellaneous valve packing leaks during warming-up and pressurization. During this period, there was an unintentional scram due to malfunction of the condenser low-vacuum trip. On June 26, half rated power (315MWt) was reached, and between June 26 and 29 power was raised in " plateaus " of 375, 435, 495, 555, 585 MWt, until full rated power of 630 MWt was achieved at 8 p.m. on June 29.

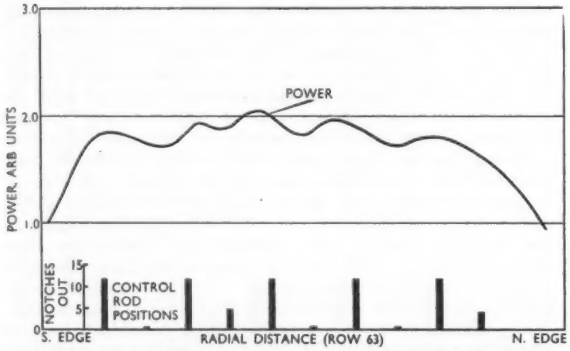
At each loading, tests were taken to demonstrate the propriety of proceeding to the next higher. These included reactor stability (rod oscillation) tests, transient tests (principally pressure disturbance) and thermal-hydraulic determinations, such as peak heat flux, and thermal burn-out margin. Other tests, not required at each power step, but carried out at appropriate intervals, included in-core wire irradiations and ion chamber calibration checks; dual-cycle load following tests; primary loop vibration and stress measurements; pressure regulator tests; recirculation pump trip-off tests; fast load rejection and turbine trips; reactor isolation; maximum primary steam flows; radiochemical and radiation surveys.



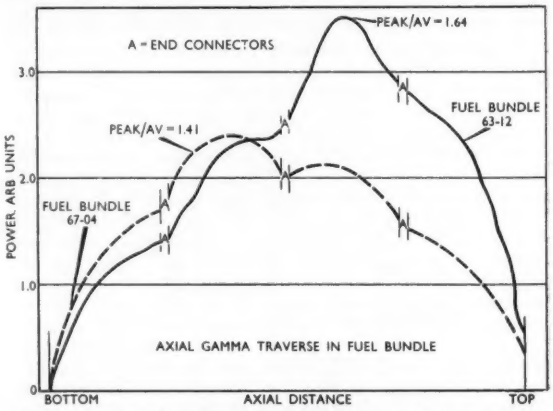
Power generation system.

Test Results

Summarizing the consolidated test and data obtained, the overall picture which emerges appears highly satisfactory. The plant was loaded, for example, to an electrical load of 194.95 MW gross (186.09 MW net) with a net thermal efficiency of 30% using high-accuracy instruments for the tests. Measurements by accelerometer and strain gauge on the steam risers showed a vibration which was hardly detectable (about 0.02 *g*) and insignificant bending stresses (less than 200 p.s.i.) even under transient conditions.



Radial traverse of core representing channel power distribution at 50% rated power. Data based on gamma probing technique.



Two typical axial power shapes based on a wire irradiation technique at 50% power rating.

Dresden's main control room.

Power Distribution Measurements

Experience gained in the interpretation of the flux monitoring equipment (by correlation with results of gamma probing during shutdown) was found useful in measuring and shaping the core-power distribution, and it was found possible to obtain peaking factors equal to or better than those obtained at half power. At 630 MWt, a peak local heat flux of 252,000 B.t.u./ft²,h was measured, compared with the figure of 280,000 B.t.u./ft²,h, allowable by licence. The minimum burn-out safety margin, also, was found to be 2.7 as against the permitted value of 2.0.

Stability Tests

Safety rod oscillation tests with periods varying from 3.4-60 sec, as carried out during the previous trials, showed no evidence of approach to unstable reactor conditions. Spectrum examination of ion chamber noise revealed no pronounced frequency peaking, such as would be expected if unstable conditions were being approached. At the higher levels, a slight peaking was noted in the 5-10 sec period range; this was not considered significant. Riser and channel flow pulsation measurements were very small, and flow shifting between risers was very slight.

Transient Tests

In addition to the pressure regulator tests carried out at each plateau, a number of severe system transients were imposed at various points in the programme.

By-pass valves. One of the eight turbine by-pass valves rapidly opened and reclosed (as would be the case during routine operation) resulted in a slight momentary fall in reactor flux and a rise of 6% when reclosed. The pressure regulator held pressure to within 4 p.s.i. of the set point.

Secondary steam valves. This, an unplanned occurrence, was caused by high water level in a steam generator, causing closing of the steam valves with the reactor operating at about 420 MWt. Reactor power decreased smoothly as sub-cooling gradually decreased. The steam valves were reset in 2 min, and the steam generator brought back to initial output in 10 min. Throughout, the flux followed the transient quite smoothly, and pressure held within 2 p.s.i., during the whole transient. Primary steam flow was virtually undisturbed.

Recirculating pumps. Two main coolant pumps were tripped at 500 MWt, giving a reduction in core flow of about 25% and a decrease in reactor power of about 17%. The flux fell off rather sharply, and gradually recovered; the undershoot was less than anticipated. The control rods were then withdrawn to the pattern for 630 MWt for normal operation and the reactor continued to operate satisfactorily, with burn-out margins and peak heat fluxes within licence limitations. This demonstration appears to favour a relaxation in the regulations requiring scram in case of pump tripping; studies are being made for reporting to the AEC.

Turbine trip. Under conditions of relatively high primary flow, and low secondary flow, the turbine was tripped at a reactor power of 450 MWt. The reactor pressure rise was higher than expected (16 p.s.i. in about ½ sec) producing a sharp neutron flux rise which returned to its original value in about 1 sec. Since the steam conditions gave core voids and void reactivity higher than normal, the flux rise was greater than would normally occur. Nevertheless, extrapolation shows that at present speeds of by-pass valve opening, flux peaks of 135% initial value may occur at rated loads; this would cause scram even though, with the short duration, there is no significant temperature rise. While this presents no hazard, it is undesirable, and studies are being made towards speeding-up by-pass operation.

Reactor Isolation. This consists of tripping the turbine stop and by-pass valves, and scramming the reactor, and virtually isolating it. Conditions were less severe than expected, the initial pressure rise being only 25 p.s.i.; insufficient to operate the relief valves or the emergency condenser. Following the initial peak, pressure dropped off and then rose again; the by-pass valves were opened after about 3 min, to prevent operation of the emergency condenser from the slow rise in pressure caused by decay heat.

Dual Cycle Performance. The characteristics of the dual cycle were investigated at various loads, and with pick-ups and load dropping at various values. Performance largely followed predictions, the major departure being in the presence of relatively larger ratio of Doppler reactivity to void reactivity, causing less response of primary steam flow to increase in secondary steam flow, than anticipated. The effect of this is not, however, significant, and the system provides good load following from 55-100% rated power.

Recirculation Loops. Coolant flow with all four pumps in operation is about 10% higher than the design value of 25.6×10^6 lb/h so that, with the by-pass flow (since modifying the guide tubes) about 8%, instead of the 5% design figure, the coolant flow is about 7% high. No account has been taken of this in calculating the burn-out values, etc., during the test programme. With all four pumps out of action, natural circulation flows of about 50% of normal have been obtained.

Radiation Levels

Radiation levels at full power are given in Table 1, and it appears that the design figures were quite conservative. Activity in plant gases was consistently low, the principal activity being N^{13} with a half-life of 10.5 min, the amount being about 1,800μc/sec at full power. Fission gas release was very low, about 100μc/sec at full power; all measurements being before hold-up prior to stack discharge. (For comparison, the licence limit for discharge of Xe and Kr is 700,000μc/sec.)

A typical analysis of the reactor water showed less than 5×10^{-5} μc/cm³ of iodine isotopes with significant half-lives (I^{131}—I^{135}).

General

To assess system margins, a test was made for maximum primary steam production. Secondary steam and reactor power levels were reduced, to remain within licence limitations. A primary steam load of 137% normal was produced, without difficulty, the limitation being the capacity of the feed pumps. Low water level, in fact, caused the termination of the test by scramming the reactor. This, incidentally, was the only "real" scram of the test period, the others being the planned isolation test referred to, and a number of spurious scrams from causes which have since been rectified. Throughout the test period, the plant has been operated in conformity with the licence.

In addition to the modifications previously mentioned, minor alterations have been made to the sodium pentaborate poison injection system, to speed up the response time under cold conditions.

Table 1. Radiation levels at rated plant output

Location	Radiation	
	Neutron mrem/h	Gamma mr/h
Near air ejector		500
Feedwater heaters (four stages)		<0.5–225
Condensate demineralizers (end of cycle)		100
Feedwater pumps		<0.5
Turbine H.P. casing		300
Turbine H.P. front standard		19
Generator collector rings		0.3
Turbine building laydown area		1–4
Condenser (tube sheet)		5
Condenser (below hotwell)		11
Near primary steam line		700
Outside reactor shield (primary and secondary)	<0.5	<0.5
Above reactor top shield	40	175
Below reactor bottom	2,000	2,000
Steam generator compartment (operating)	<20	1,500

Fig. 1.—PM-1, a 1,000-kW nuclear power plant being built by The Martin Company for an Air Force station in a mountainous section of the western U.S., is designed for air shipment in 16 packages similar to the one shown in the photograph.

Remote Military Power Plants

By FRED HITTMAN
(Manager, Nuclear Powerplant Dept., The Martin Company)

THE military requirements for portable nuclear power plants operating in remote locations dictate design and performance characteristics which are different from and more stringent than those for central station nuclear plants. At least five such major performance and design requirements might needs be considered.

1. *Portability.* Portable nuclear power plants designed and built for use in remote regions, like the PM-1[1], PM-2A[2], and PM-3A[3], are designed to be transportable by air in aircraft that can land on the strips available at such remote sites as McMurdo Sound in the Antarctic. This places total weight, size, and centre of gravity requirements on the plant packages themselves, as well as stringent " G " loading requirements on the equipment and equipment mountings within each package. Based on the C-130A aircraft, these limitations must be observed:

Package size: Size not to exceed 8 ft 8 in. × 8 ft 8 in. × 30 ft.
Package weight: Weight not to exceed 30,000 lb total.
" G " loading requirements: (a) Limit loads (i.e., equipment must still be operable): 3*g* fore and aft, vertical and downward; 1½*g* laterally. (b) Ultimate loads (i.e., in case of a crash landing, equipment mounts must not yield): 8*g* forward, 4½*g* downward, 1½*g* laterally.

In the design development work on the PM-1 a full-scale package simulating all equipment and equipment mounts was constructed, statically and dynamically tested, and then loaded and test flown in a C-130A aircraft (Fig. 1). One of the 27 packages which make up PM-2A was also delivered to Greenland by air for demonstration purposes. PM-3A will meet the same design criteria, but is expected to be delivered to McMurdo Sound by ship.

2. *Prepackaging.* As the environments in which these plants are used allow extremely short construction seasons, and the personnel available for installation and test of the plants is limited, the erection time for portable nuclear power plants must not exceed 60-90 days and a minimum of field welding is permitted. This means that the entire plant must be pre-assembled and tested at the fabrication plant as far as possible and that the equipment must be fitted into the smallest possible number of packages.

Miniaturization therefore becomes important and consequently engineering costs tend to be high. Even the smallest piping and valving must be fully engineered and cannot be field run. All equipment is mounted on rigid steel skids which can be placed directly on foundations and interconnected at the site.

In the case of the PM-1, for example, this results in a plant consisting of 16 air transportable packages, including a fully pre-assembled decontamination and maintenance facility, modular evaporative waste disposal system and a fresh and spent fuel storage facility. Fig. 2 shows the PM-1 as it will appear when installed at the Sundance Wyoming Air Force Station. Fig. 3 shows two of the skids of the PM-2A plant, presently being installed at Camp Century.

3. *Reliability.* Because portable military reactors are usually operated in locations where the success of the mission may depend heavily on the continued production of power, great reliability is essential. The necessity for reliability cannot be overemphasized and results in strict design criteria governing allowable down time, time for refuelling, core life, etc. Some typical design criteria are shown in Table 1.

TABLE 1.—Typical Design Criteria.

a.	Equipment Design Life	20 years.
b.	Total Down Time permitted per year	21 days.
c.	Refuelling Time	4 days maximum.
d.	Core Life	2 years at full power.
e.	Climatic Range:	
	wind load	100 m.p.h.
	temperature range	125°F to −60°F.
	snow loads	30 lb/ft².
f.	Control Components	Fail-safe, solid state type with self checking circuitry that locates and annunciates location of faulty instrument modules for easy and rapid repair.
g.	Operating Crew Size	*Minimum.* Not to exceed two operators per shift.
h.	Self-Start-up and Emergency Power Capability.	Plant must have an auxiliary diesel and emergency power system to allow independent start-up after shutdown and provide heat and power during maintenance periods.

In addition, as the sites at which these plants are used normally do not have sufficient cooling water for heat rejection, plant cycle heat must be rejected to the atmosphere. A considerable additional difficulty is posed by this requirement because the temperatures at these sites are also very low (to −60°F) and can result in serious freeze-up problems. In the PM-1 plant, heat dissipation is accomplished by direct air-steam condensers designed and developed to produce the required amount of cooling over an ambient air temperature range of 70°F to −60°F. Fig. 4 shows a model of the PM-1 condenser undergoing testing at −65°F in a climatic chamber. In the PM-2A plant an ethylene glycol water solution is used as an intermediate condensing fluid, with the final heat rejection taking place in an ethylene glycol to air heat exchanger.

In order to ensure continued operation, all failure-prone components are duplicated. Those that are not are so arranged that they can be replaced quickly by spares provided with the plant, using specially designed handling tools.

4. *Power Output Requirements and Economics.* The power output requirements for portable military plants are quite nominal and in the range normally provided by diesel engines when conventional fuel is readily available. The power and/or process heat outputs of the PM-1, PM-2A, and PM-3A plants are shown in Table 2.

TABLE 2

Plant	Reactor Thermal Output	Core Life	Plant Electrical Output	Plant Heating Output
PM-1	9.37 MW	18.74 MW yr	1,000 kW	7×10^6 B.t.u./h
PM-2A[4]	10 MW	10 MW yr	1,500 kW	1×10^6 B.t.u./h
PM-3A	10.4 MW	18.74 MW yr	1,500 kW	$.5\times10^6$ B.t.u./h

The economics surrounding use of such plants is intimately tied to the site at which they are used. Their freedom from the logistical problem of continual fuel supply can easily overshadow their higher initial cost when compared with conventional plants. In locations such as McMurdo Sound, Antarctica (the PM-3A site), where conventional fuel supplies are very costly, it has been shown that over the life of the plant the nuclear plant is much more economical than its conventional competitor. A recent study[5] indicates that nuclear power costs at McMurdo Sound may be of the order of one-half the cost of conventional diesel power.

5. *Special Nuclear Requirements and Problems.* The most important requirements of these reactors are that they be sufficiently compact to meet the transportability requirements spoken of before, that they can be operated safely and reliably, and that they can be refuelled easily and safely at infrequent intervals. Because of the requirement of proven system reliability that is placed upon these systems, all field plants presently under construction are of the pressurized water type, although boiling water, gas cooled, and other more advanced systems are also under development by the U.S. Atomic Energy Commission.

Minimization of size and weight of the primary shielding and containment (if required) pose additional problems, but a brief description of the PM-1 primary loop serves to show how these problems can be solved:

The PM-1 primary loop (Fig. 2) is contained in two cylindrical tanks 8 ft 8 in. diameter and 30 ft long, constructed of high alloy steel. A third tank connected by a passageway to the reactor tank serves as an irradiated fuel storage area. A simple steel structure is used to support these tanks, the primary loop building, and the gantry crane (which is used to erect the tanks initially and for maintenance and refuelling thereafter). Local aggregate is placed around the primary loop tanks to provide secondary shielding.

The reactor tank contains the pressure vessel, nuclear controls and instrumentation, and primary shielding. In operation it is filled with water, which serves as shielding in the vertical direction. A specially designed, weight-optimized lead and water shield surrounding the pressure vessel provides radial shielding. The second large-diameter tank is operated dry and contains: (1) a vertical steam generator of Inconel-type

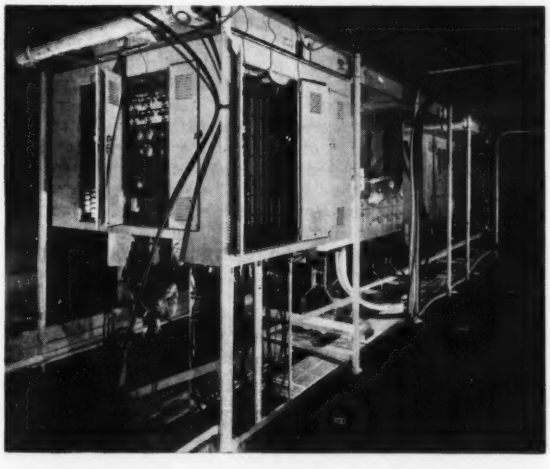


Fig. 3.—PM-2A, Alco Products installation for Camp Century, showing the control skid and switchgear package in background.

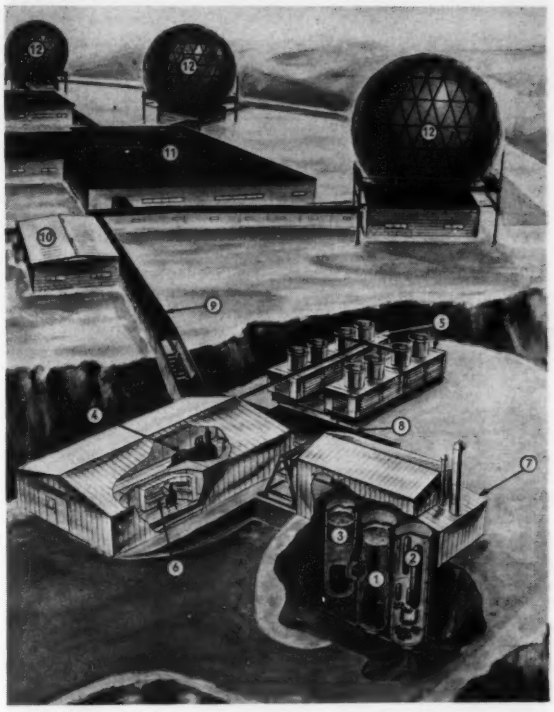


Fig. 2.—Artist's concept of a typical PM-1 installation : providing power for a remote radar installation.

KEY. 1. Reactor tank. 2. Steam generator tank. 3. Spent fuel storage tank. 4. Steam turbine and electric generator. 5. Air-steam condensers. 6. Control console. 7. Shield water cooler. 8. Decontamination and water chemistry laboratory. 9. Covered walkway. 10. Base technical supply building. 11. Base operations building. 12. Radar installations.

construction; (2) a canned motor pump; (3) the vertical pressurizer; (4) high pressure demineralizer for continuous purification of the primary loop water; (5) a low pressure demineralizer for continuous purification of the shield water in the reactor tank and the fuel storage tanks; (6) an expansion tank; (7) the 8-in. dia. 316 stainless steel primary piping.

Refuelling can be accomplished quickly and safely by removing the entire core from the reactor and transferring it by a hand-operated dolly through the flooded passage between the reactor tank and the fuel storage tank.

The PM-1 pressure vessel, which is $39\frac{3}{4}$ in. O.D. and $2\frac{1}{16}$ in. thick, is fabricated of 347 stainless steel for a 20-year life and to preclude the problems associated with materials whose nil ductility transition temperatures are raised by neutron irradiation to unacceptable levels.

The active core is a cylinder of 23.6 in. diameter and 30 in. long (Figs. 5 and 6). Six peripheral pie-shaped bundles and one central hexagonal bundle make up the core (Fig. 7). Because the reactor must be refuelled in a minimum of time, the core is designed to permit refuelling as a single unit by removal of the entire core (including control rods) in its shroud. For flexibility, refuelling by individual bundle is also possible. The central bundle can be removed and replaced without removing the pressure vessel head and is designed to permit installation of in-core instrumentation. The core contains 732 tubular fully enriched UO_2-stainless steel cermet type fuel elements 0.500 in. O.D. and 0.416 in. I.D.

Coolant at 1,300 p.s.i.a. and 2,125 gal min flows in a single pass through both the inside and outside of the fuel tubes. Local boiling is permitted in the core, which has an average and maximum heat flux of 73,000 and 290,000 B.t.u./ft² h respectively. Control is provided by six Y-shaped control rods using Eu_2O_3 stabilized against hydration by TiO_2 as poison material. The rods are actuated by simplified small diameter magnetic jacks which can be maintained without depressurizing the primary loop.

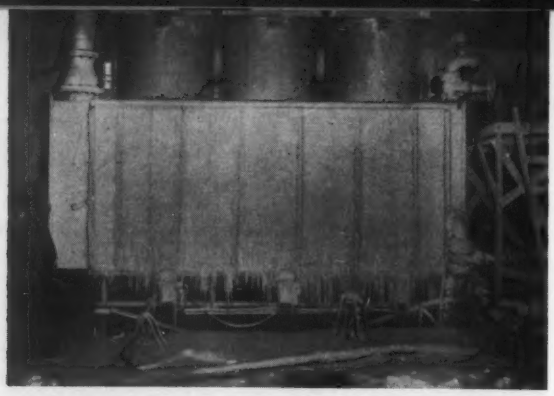


Fig. 4.—Some remote locations require operation in unusual temperature ranges. A test model of the air-steam condenser for PM-1 is checked in an environmental chamber at −65°F.

Fig. 5.—Gross flow test core for Martin's PM-1, showing details of positioning guides for Y-shaped europium oxide control rods.

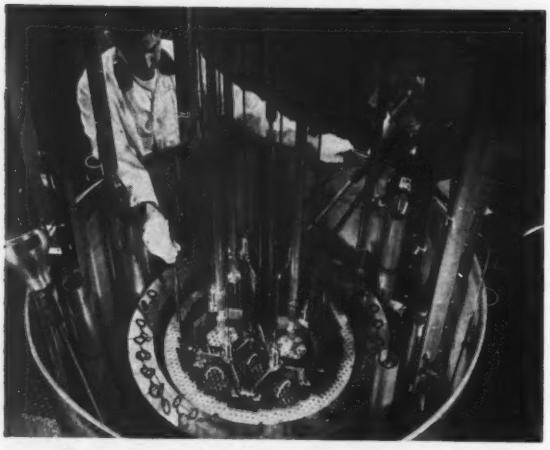
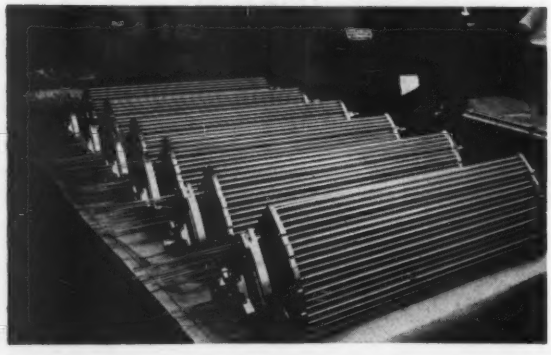


Fig. 6.—Zero power tests on the PM-1 core.

Fig. 7.—(Below) Bundles of tube elements for PM-1 gross flow tests.



Seventy-five ½ in. O.D. natural boron stainless steel alloy lumped burnable poison rods are distributed throughout the core, which has a total fuel loading of about 28 kg of U^{235}. Sufficient control margin is provided so that the reactor can be shut down with one rod stuck fully out or two in the operating position. The use of enriched fuel and lumped burnable poisons yields a compact core design without unduly limiting core life or control capability.

The negative temperature coefficient at operating conditions of approximately -2.1×10^{-4} ($\Delta k/°F$) ensures good self-controlling characteristics.

The steam generator generates 300 p.s.i.a. saturated steam which drives the turbine generator, rated at 1,250 kW(e) gross. Cycle heat ($20 \times 2 \times 10^6$ B.t.u./h) is rejected by direct air-steam condensers developed and designed for operation over a temperature range of 70°F to −65°F. Nuclear controls and instrumentation, as well as all other plant instrumentation, are centralized in one console and are of the solid state type.

Influence on Civilian Developments

It is always difficult to predict what influence one technical project will have on others, but it seems logical that the U.S. programme for portable military power could affect civilian power efforts in several ways:

(1) As the capital costs of the small portable nuclear plants are considerably less than large scale civilian plants, they might serve as inexpensive means of testing new and advanced reactor concepts which could then be utilized in large central station power plants.

(2) The technique of prefabricating and pre-assembling most portable power-plant components into modules which are interconnected at the final site might be applied to help reduce the on-site fabrication of larger central station plants. (Of course, the size and transportability of modules for larger plants would tend to limit such efforts to some degree.)

(3) Design features used to reduce the size of equipment such as pressure vessels, piping and heat exchangers, may result in capital cost savings when applied to large stationary plants. Furthermore, the use of small-volume, high-pressure containment schemes (i.e., final pressures of the order of 100-150 p.s.i.a.) could well be considered by civilian reactor builders as an alternative to the large L.P. containment shells now in general use.

(4) The use of certain special equipment and techniques which are necessary to reduce operating problems in remote areas might result in lower operating costs at civilian plants as well if they could be adapted suitably. These include simplified refuelling techniques, reliable and long-lived control rods and control rod actuators, self-checking solid state instrumentation, and automated equipment in general.

(5) For economic reasons, most civilian stations have used reactor cores fuelled with natural or slightly enriched uranium in the form of metallic alloys or ceramics, while the cores for portable military reactors to date have concentrated on the highly enriched UO_2-cermet metallurgy in order to achieve compact core geometries and high fuel burn-up.

Although future military reactors may benefit somewhat in economics by exploring the design of compact, low-enriched UO_2-ceramic pellet type cores, the designers of civilian reactors may profit equally by examining the use of highly enriched cermet fuel elements as seeds for large cores.

To a larger degree, the "packaged" power plants developed by the United States for remote areas represent a distinct field in themselves. Despite their unusual problems and sometimes even more unusual solutions, however, the technology they are producing makes them an important factor in the general growth of the nuclear industry and nuclear engineering.

References

1. PM-1 (Portable Medium Power Reactor No. 1) built by Martin for U.S. AEC to be installed near Sundance, Wyoming.
2. PM-2A (Portable Medium Power Reactor No. 2A) built by ALCO for U.S. Army for installation at Camp Century.
3. PM-3A (Portable Medium Power Reactor No. 3A) being built by Martin for U.S. AEC for installation at McMurdo Sound, Antarctica.
4. "Evaluation of the Army Package Power Reactor Family," by J. G. Gallagher, Nuclear Engineering and Science Conference, Cleveland, Ohio, April, 1959.
5. NYO-2947 Study of Remote Military Applications. Report No. 11. U.S. Naval Air Facility—McMurdo Sound, Antarctica, for U.S. AEC, NYOO. February, 1960, by Kaiser Engineers, Oakland, California.

NUCLEAR ENGINEERING

AEC-Sponsored CTR Research

By Dr. F. L. RIBE,
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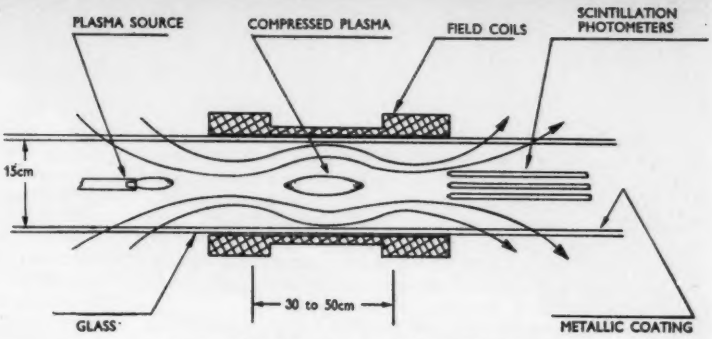


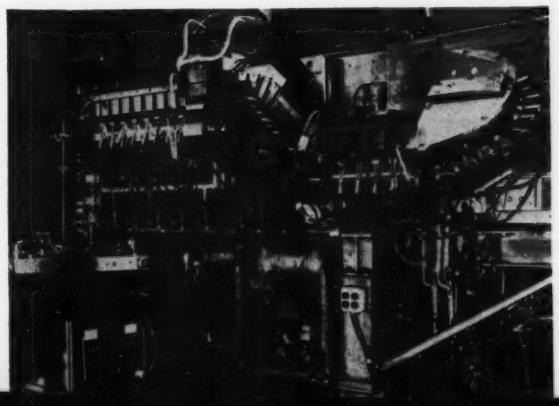
Fig. 1.—Magnetic mirror machine, Table-Top.

IN attempting a survey of the experimental work of the United States controlled thermonuclear research effort, one must drastically limit the choice of material in an article of reasonable length. Therefore, in the following summary, many significant results have been omitted, and those results which are reported have to do largely with well-established experiments. A notable omission is the pinch work being carried out at Los Alamos, General Atomic in San Diego, and at the Lawrence Radiation Laboratory in Livermore. As has been extensively reported, the most striking general result is the appearance of plasma energy losses in the form of ultraviolet radiation which are a thousandfold greater than those to be expected from a pure deuterium plasma. One may also briefly mention the rotating plasma experiments at Livermore and Los Alamos in which the plasma energy limitation appears to be brought about by collisions of the rotating charged particles with neutral background atoms. We shall finally discuss one newly conceived type of experiment.

The Mirror Machine

At Livermore, considerable evidence has been obtained to demonstrate the existence of a plasma which is indeed stable under conditions for which simple theory would indicate instability. In addition plasma diffusion across magnetic field lines has been shown not to exceed the classical value within experimental error—a fact which augurs well for the success of plasma containment generally. The apparatus is a magnetic mirror machine, Table-Top (Fig. 1), in which a burst of low density hydrogen plasma is deposited in the machine and then adiabatically compressed and heated by the slowly rising magnetic field. The energy of the resulting plasma resides mostly in the electrons whose final temperature is of the order of 20 keV. The plasma density is roughly 10^{14} particles/cm^3, and the ratio β of plasma pressure to magnetic pressure (or energy density) is a few per cent. For containing magnetic fields of the order of 20 kgauss and a magnetic mirror whose field is twice as great at the ends as in the middle (mirror ratio=2) the plasma is contained for the order of 10 m sec. As the theoretical time for growth of hydromagnetic flute instabilities is only about 0.1 μ sec we have a demonstration of plasma stability. The diffusion rate of plasma across the magnetic field is found to be 0 ± 40 cm/sec, as compared with the classical computed value of 0.4 cm/sec. By deliberately augmenting the velocity of the electrons around the magnetic field lines, relative to that along them, it is found that a new type of instability—the so-called mirror instability—can be induced.

Fig. 2.—(Below) B-3 Stellarator.



The Stellarator

At Princeton the experimental approach is that of the Stellarator, in which a heated plasma is contained by a longitudinal magnetic field. Both the plasma and the field lines follow a vacuum chamber which is shaped roughly like an oval race track. However, it is necessary either to twist the oval into a figure eight as in the case of the B-3 Stellarator (Fig. 2) or to apply helical windings to provide a device in which a given magnetic line does not return upon itself ("Etude" Stellarator) in order that the plasma does not drift out of the vacuum chamber.

It has been evident for some time that ultra-high vacuum is necessary for Stellarator-type machines in which plasmas are held for long periods ($\sim$ 100 μ sec) by means of magnetic fields. In the past, entry of impurities from substances adsorbed on the vacuum-chamber walls has been so severe that the number of impurity ions in the discharge often exceeded the number of hydrogen ions. A clear-cut improvement has been observed as a result of elaborate vacuum clean-up procedures which are at present unique to the Stellarator project. Ultra-high base vacua in the region of 10^{-9} mm Hg are obtained in large systems baked out at 450°C and additionally cleaned by approximately 1,000 discharges before plasma measurements are made. In clean Stellarator B-3 and "Etude" systems impurity ion contamination has been reduced to a few per cent. The process of "pump-out," whereby the plasma is suddenly lost to the wall, is now observed reproducibly in helium as well as hydrogen. The pump-out process appears to be associated with plasma instabilities arising from directed particle motions produced by the longitudinal currents whose joule heating is thus far the principal plasma-heating means.

Fast Magnetic Compression Devices

At Los Alamos and the Naval Research Laboratory, considerable evidence exists that both the ions and electrons of dense deuterium plasmas can be heated to temperatures of the order of keV by the application of rapidly rising, large magnetic fields to deuterium gas. The mirror devices employed differ somewhat in speed and dimensions but appear to involve similar mechanisms for plasma heating in which an initial violent plasma "shock" is followed by adiabatic compression in magnetic fields of the order of 100 kgauss. A photograph of the Los Alamos device, Scylla II, is shown in Fig. 3. A massive beryllium-copper compression coil can be seen at the

Fig. 3.—(Right) Scylla II, the Los Alamos device.

right, clamped at its current input by two beams. At the left can be seen five of the 10 100-kV capacitors with associated spark gaps on top which furnish the fast rising magnetic field. The two large vertical cubicles at the centre contain capacitors which furnish a slower compression field when needed. The régime of these devices is quite different from that of the much slower Table-Top machine, as the plasma density and β-value are here much greater, while the compression times and plasma dimensions are much smaller.

In the NRL device the compression field rises to its maximum value in a time of the order of 5 μ sec. The plasma is observed as a function of time by means of a moving image camera which shows that a plasma of density $\sim 10^{17}$ particles/cm^3 exists without gross instability for a few microseconds, accompanied by intense bursts of neutrons from the d-d nuclear reaction whose duration corresponds closely to that of the compressed plasma. Measurements have been made which show that the magnetic containment of these plasmas is determined by considerations unique to high density plasmas (again in contrast to the Table-Top machine). The plasma containment time τ is found to be a linear function of the product of the machine's length L and its mirror ratio R over the limits $1<\tau<7$ μ sec and $7<LR<75$ cm.

In the Los Alamos device, Scylla, the compression field rises in approximately 1 μ sec. Work has concentrated on measuring the plasma properties such as its size, plasma density, electron temperature, and ion temperature by means of various radiations which the plasma emits. The nuclear reaction particles (neutrons, protons, and tritons) from the d-d reactions in the plasma are used to determine its size and deuteron ion temperature. Soft X-rays and visible and ultraviolet radiation determine the electron temperature, plasma density, and degree of impurity. There is good agreement between the plasma deuteron temperature (1.4 keV) deduced from the yield of d-d neutrons and that deduced more directly from the measured spread of deuteron energies in the plasma. In both the NRL and Los Alamos devices the necessity (discovered at NRL) of trapping reversed magnetic field inside the plasma in order to obtain energetic ions is established, and work is being done to determine the mechanisms involved. There are strong indications that both devices produce high-β plasmas, but essential work remains to be done to establish this fact and to determine its consequences.

The DCX Machine

At Oak Ridge the DCX principle is used to build up an energetic, low density plasma by injecting a beam of hydrogen molecular ions across a magnetic field along which there is a carbon arc. The molecular ions are broken up into atomic ions by the arc, and the atomic hydrogen ions are trapped into orbits and contained in a mirror magnetic field which is supplied by d.c. power. The object of the experiment is to achieve "burnout," whereby the population of background neutral atoms is greatly reduced by virtue of their ionization by the trapped ions, thus permitting the hot ion density to build up to large values. In the present machine, DCX-1, a hydrogen plasma has indeed been built up to an ion density of 10^{10}/cm^3 with a lifetime of approximately 10 m sec. The reasons for plasma loss and lack of burnout are understood and accounted for experimentally by measuring the collisional losses arising in the arc and neutral background gas. No unknown phenomena, such as instabilities, are found to affect the machine. A new machine DCX-2 (Fig. 4) is being designed in which the deleterious effects of the arc are eliminated by allowing the plasma first to build up through collisions of beam molecules with the arc which is then turned off. Further beam break-up and trapping will then take place by means of collisions of the beam molecules with ions in the hydrogen plasma.

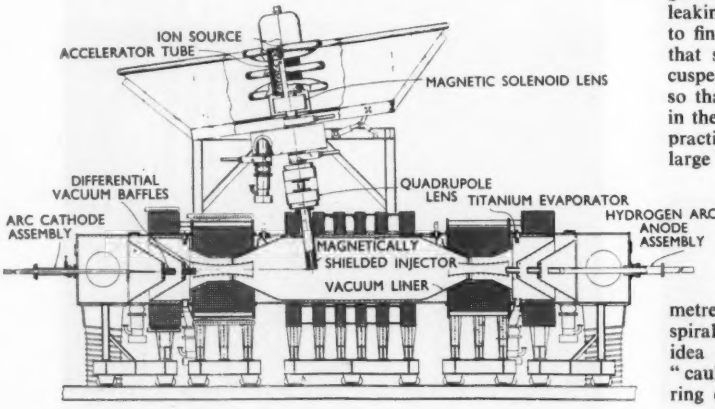


Fig. 4.—DCX-2 is being designed at Oak Ridge.

Cusped Magnetic Fields

A relatively new approach to the thermonuclear problem involves injecting a dense stream of energetic plasma from a gun into a magnetic bottle in such a way as to "inflate" it. In the entropy trapping concept a fast jet of relatively cold plasma is projected axially into a picket fence or cusped magnetic-field geometry. The plasma exerts a dynamic pressure on the magnetic field, and this is made great enough to force the magnetic lines apart, allowing the fast jet to enter the central region of the confinement geometry. The action of forcing its way in, together with scattering of the beam off the inside magnetic walls, randomizes the individual particle motions so that on turning the jet off, the inside plasma cannot exert enough pressure to force its way out and it is consequently trapped. This method, originated at Los Alamos, is being studied also at General Atomic in San Diego and at Livermore.

Various types of plasma guns are used; a Los Alamos gun has achieved a stream of hydrogen plasma of some 5×10^{16} ions having an average energy of the order of 10 keV. In the General Atomic experiment the plasma is observed to open the input throat of the cusp and to be detained inside for about 40 μ sec with a number density greater than 10^{15} particles/cm^3. In the Los Alamos experiment the plasma is observed to force a passage through the input of the cusp under certain conditions of density and velocity and to remain in the cusp for the order of 40 μ sec. One expects theoretically that if the magnetic field at the input to the cusp is made sufficiently strong its pressure should lead to the exclusion of the plasma stream. This effect has so far not been observed in the cusp experiments, in spite of extensive experiments at Livermore.

The theoretical prediction by Trubnikov and Kudryavtsev of the U.S.S.R., of strong radiation from a magnetized plasma at the temperatures needed for a thermonuclear reactor, revealed at the 1958 Geneva Atoms for Peace Conference, has been confirmed theoretically in this country. The predictions make it harder to construct a d-T reactor and extremely difficult to make a d-d reactor using a magnetized, i.e. low-β, plasma. One can get around this difficulty, in principle, to a large extent by having a non-magnetized plasma, i.e. β=unity. However, of all the magnetic confining devices yet considered, only one has enough hydromagnetic stability to confine such a plasma—namely the picket fence or cusped geometry. For this reason, Los Alamos research is concentrating increasingly on this system of confinement. The disadvantage of the picket fence geometry is that the long cusped regions add greatly to the leakiness, and the workers at Los Alamos have been attempting to find some way out of this difficulty. There is a possibility that such a way has been found, based on the picket fence cusped geometries in which the line cusps are joined together so that leakage from them is defeated. This new geometry is in the idea stage and its possibilities are being analysed. In its practical embodiment, it seems likely that it might consist of a large straight solenoid some metres in diameter, and many metres in length, having a large, magnetically supported inner spiral winding. Experimental studies of the feasibility of this idea will take the form, as an intermediate step, of the "caulked picket fence" which uses a single current-carrying ring conductor supported inside a magnetic solenoid.

N.S. SAVANNAH

THE commissioning of the N.S. *Savannah* will mark the true commencement of nuclear merchant propulsion. Although not the first vessel to be propelled by nuclear energy, being antedated by a fleet of submarines and the icebreaker *Lenin*, she will be the first attempt at nuclear propulsion of a commercial vessel. *Savannah* is more than a floating laboratory or a costly gesture in the field of international prestige. In the nuclear field she will provide the answer to a great many technical problems; in the field of international shipping and insurance she will resolve into tangible form a problem that, up to now, has been argued on the basis of an equation with too many unknowns.

Savannah's keel was laid on May 22, 1958, at Camden, New Jersey, and she was launched, by Mrs. Eisenhower, on July 21, 1959. She is 595 ft 6 in. in overall length, 78 ft moulded beam, draws 29 ft 6 in. fully loaded, and has a displacement of 22,000 tons. With a shaft horsepower of 20,000 (22,000 max.) she has a normal speed of 21 knots, and an estimated cruising range of 300,000 nautical miles without refuelling. She will have a cargo capacity of 9,400 tons deadweight, and accommodation for 60 passengers.

Propulsion is by means of a De Laval turbine driving a single propeller through reduction gearing; provision is made for emergency operation by a 750 h.p. electric motor. There are two 1,500 kW turbo-generators for the ship's electrical load, and two 750 kW diesel stand-by generating sets.

Steam is provided from a reactor of the PWR type with a maximum operating power of 69 MWt through two heat exchangers.

The *Savannah* is the joint responsibility of the Maritime Administration of the U.S. Department of Commerce and the U.S. AEC. Designed by George E. Sharp, Inc., she was built by the New York Shipbuilding Corporation, at Camden, New Jersey. The prime contract for the propulsion machinery was awarded to The Babcock and Wilcox Company, with the De Laval Steam Turbine Company as sub-contractor for the turbines, reduction gearing, and other heavy plant.

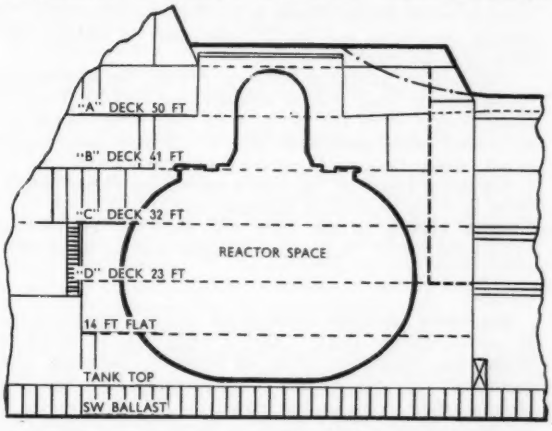
A preliminary description of the *Savannah* appeared in *Nuclear Engineering* for May, 1959, p. 197.

(Right) Diagram showing location of reactor. (Below) Sections of the inner bottom showing "egg crate" construction.

HULL

Savannah has three complete decks, and 10 main transverse bulkheads; i.e., 11 compartments. She is transverse-framed, except for the inner bottom, which is a combination of transverse and longitudinal framing. In the reactor area, the bottom is specially stiffened, an "egg-crate" construction being used, with transverse floors at every frame.

The reactor compartment is located approximately amidships, between bulkheads 5 and 6; the propulsion machinery being immediately aft, between bulkheads 6 and 7. The reactor compartment has heavy longitudinal collision bulkheads



21 ft 9 in. from the centre line; outboard of these. decks B, C, and D have extra-heavy plating, continuously welded to the deck beams. Inboard of these are the collision mats, extending vertically from C deck to the 14-ft flat below D deck, and 35 ft long. This mat is made up of alternate layers of 1 in. steel and 3 in. redwood, the total thickness being 24 in.

REACTOR

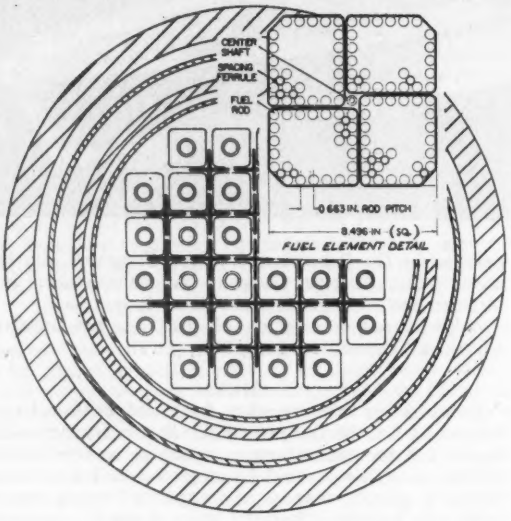
The reactor is a pressurized light water unit with two heat exchanger loops. The reactor, pressurizer, heat exchangers (steam generators), coolant circulating pumps, and a number of auxiliaries, are located within the main containment, which has its own closed-circuit air conditioning system. Some of the low-pressure auxiliaries, such as the primary water demineralizers, etc., are located outside the containment.

Core

The core, in cross-section, can be regarded as of cruciform shape, as a square with notched corners or as approximately

circular. It consists of 32 fuel elements arranged in six rows of 4-6-6-6-6-4, in an "egg-crate" structure of stainless steel with a lattice pitch of 9.7 in., secured between upper and lower grid plates, and baffled so that the coolant makes three vertical passes in traversing the core.

The fuel elements are 8.5 in. square, and there are 164 fuel rods in each element, arranged in four bundles of 41 rods.



Cross-section of pressure vessel and core showing detail of fuel element.

Each rod is 0.5 in. overall diameter, consisting of enriched uranium oxide pellets canned in 0.035 in. stainless steel. The average enrichment is 4.4% U^{235}, the inner 16 elements being 4.2%, and the outer 4.6%, enriched.

The rods are spaced at 0.663 in. centres, the spacing being maintained by small tubular stainless steel ferrules, brazed on about every 8 in. of the length of the rod. This spacing gives an overall fuel-to-water ratio of 0.76. Helium under pressure is used inside the rods to ensure good heat transfer.

The active length of the core is 66 in.; the overall length is 90.24 in. The equivalent diameter is 62.06 in.

Initial loading of the core will be 6,788 kg U^{238} and 312.4 kg of U^{235}. The conversion ratio will be about 0.4.

Control

The excess reactivity required for the cold clean core will be approximately 11.2% $\Delta k/k$; made up of 3.2% for temperature coefficient from 68°F-508°F (including power coefficient); 2% for Xe and Sm poisoning; 1.3% for power Doppler effect (zero to full power); 4.7% for burn-up and isotope build-up. Total reactivity vested in the control rods is 14-18% $\Delta k/k$.

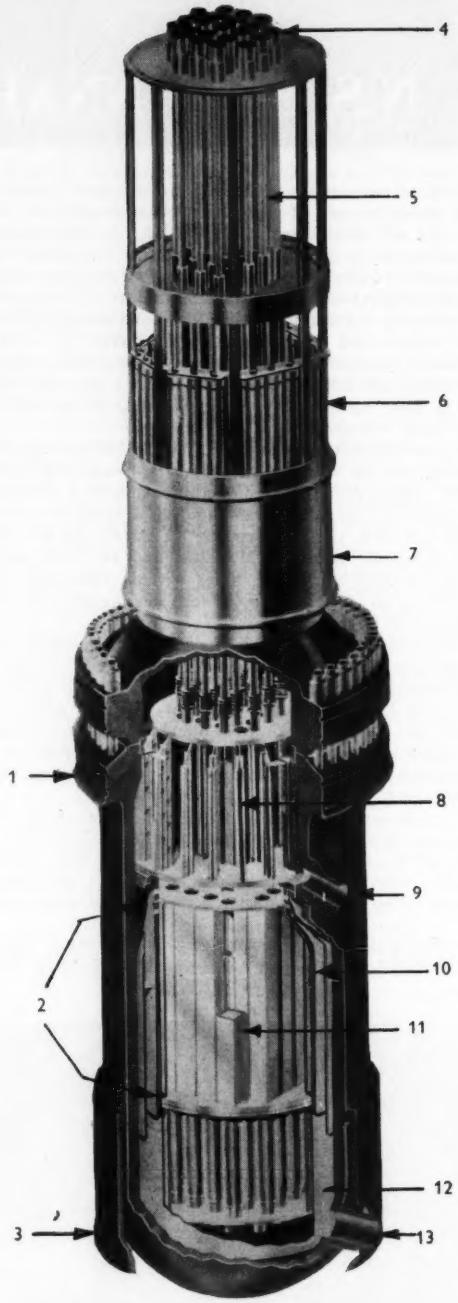
There are 21 cruciform control rods, 8 in. square and 66 in. effective length. The thickness of the section is $\frac{1}{4}$ in., made up of a $\frac{1}{16}$ in. plate of boron stainless steel clad both sides with $\frac{3}{32}$ in. stainless steel.

The control rod operation is by electromechanical means, with the addition of hydraulic cylinders for scramming. The lack of a permanent vertical and the fact that, in heavy weather, g might have a variable—or even negative—value, makes gravity operation of rods quite unreliable, and leads to the most obvious departure from land-based reactor designs; the control mechanisms are the subject of a separate article.

Primary Coolant Circuit

In the primary circuit, 8,000,000 lb/h of demineralized water is circulated in two loops, by four pumps, through two heat exchangers or steam generators. The pumps are of the vertical canned rotor type, designed for a flow of 5,000 gal/min at 495°F, against a head of 70 p.s.i. The motors, each rated at 230 kW, are designed for two-speed operation so that, if desirable, they may be run at half speed to remove decay heat after shut-down.

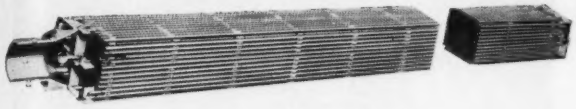
Main piping for the coolant circuit is $12\frac{3}{16}$ in. i.d., and the branch for each pump is $8\frac{3}{16}$ in. All primary piping is made from Type 304 stainless steel forgings. Each primary loop contains two motor-operated valves for isolation from the reactor, and each pump circuit has, downstream of the pump, a check valve to prevent reverse flow in case of pump failure;



Cutaway view of reactor: (1) flange (2) effective core height (3) support skirt (4) motor drives (5) hydraulic cylinders (6) lead-screw (7) gland seal closure (8) control rod (9) water outlet (10) thermal shields (11) fuel elements (12) flow baffle (13) water inlet.

these valves have small holes in the disc, for temperature equalization.

Saturation pressure of 1,750 p.s.i. is maintained in the primary circuit by means of a pressurizer, in which, following normal practice, a small steam volume is kept at constant pressure by energizing immersion heaters or water sprays.



Cutaway view of fuel element.

Pressure Vessel

The reactor pressure vessel is of carbon steel clad with stainless steel and is 26 ft 6 in. high, 8 ft 2 in. inside diameter, and 6 in. thick. It was designed for 2,000 p.s.i. and 650°F. The removable head is secured by 48 studs, 5 in. diameter, Sealing is by double gaskets and a seal weld between the flanges.

Heat Exchangers

The heat exchangers, or steam generators, are of the shell-and-tube type, both shell and tubes being in the form of a U. The primary water is contained in the tubes (800 ¾ in. diameter stainless steel).

On the secondary side, the shell connects with a separate steam and water drum, by means of 13 risers and eight down-comers, to give natural circulation in any attitude of the ship. The drum contains cyclone separators and steam scrubbers, and supplies steam at approximately 0.25% moisture content. The total steam generation for both heat exchangers is 242,000 lb/h at normal load, the steam conditions being 490 p.s.i.a., dry saturated. The feed temperature is 347°F.

Reactor Auxiliary Plant

In addition to the coolant and heat exchange circuits there are, of course, several auxiliary circuits required for normal operation, as well as in emergency.

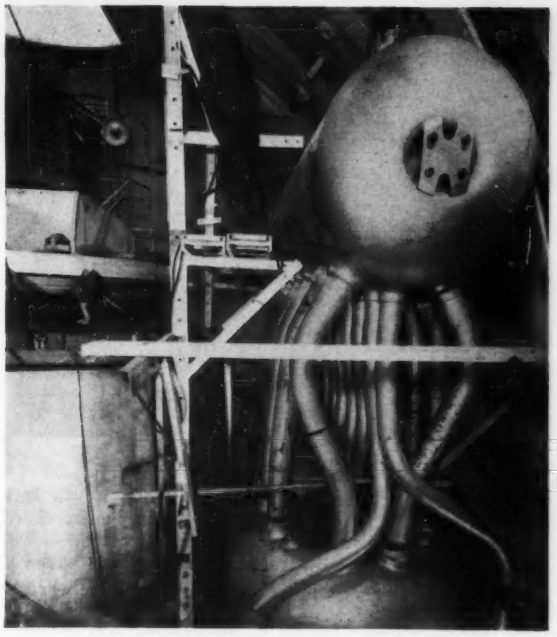
Primary Purification System. Approximately 20 gal/min of the primary water is cooled and de-pressurized to 110°F and 50 p.s.i., and passed through filters and demineralizers before being returned to the primary circuit. There are three demineralizers, each having an estimated life of 50 days so that, used in turn, replacement of the beds should only be necessary after 150 days of operation. There is also a **Hydrogen Injection System** which ensures that the minimum concentration of dissolved hydrogen in the primary loop is 20 cm^3/litre to ensure scavenging of oxygen liberated by radiolytic action.

Although the purified water may be returned to the primary system direct, through a charging line and regulating valve, it is normally returned through the buffer seals.

Buffer Seal System. The control rods are sealed where they pass out of the vessel by back-pressure seals, to prevent coolant leakage, and the purified primary water is returned through these seals by three charge pumps, only one of which will normally be in operation when the seals are new and leakage is a minimum; two pumps may be necessary when the seals are nearing the end of their working life. The stand-by pump starts automatically in the event of a failure of the running pump or increased flow requirements.

Component Waste and Drain System. All drainage that might be radioactive is collected by a system having two pumps, a containment drain tank, four waste storage tanks and a laboratory waste tank, the total storage volume being 1,320.5 ft^3. Provision is made for sampling the contents of any tank, after which the contents may be pumped to either ship storage tanks or to dock facilities. Normally, the waste collected will be of very low activity.

Gaseous Waste System. Radioactive gases released by a fuel cladding failure will be recovered from a stripping column on the buffer seal surge tank and collected, along with any gaseous discharge from any other components. Oxygen



Heat exchanger installed in containment.

present is recombined with hydrogen, and fission product gases are adsorbed on activated charcoal at a low temperature (−280°F). Any uncondensable gases are collected in a common manifold and—if environmental conditions are favourable—will be diluted and discharged to the stack which forms the radio mast.

Sampling System. An extensive sampling system is provided, which enables continuous monitoring of the primary water for fission products and representative samples for purity testing. It also allows samples to be obtained from the laboratory waste tank, the high and low activity tanks and the inner bottom storage tanks.

Primary Relief System. Relief valves at various points in the primary system vent into the effluent condensing tank, and the condensed liquid is handled by the component waste and drain system. Non-condensing gas is passed to the gaseous waste disposal system. If, in emergency, the discharge exceeds the capacity of the effluent condensing system, a relief valve allows discharge direct into the containment vessel.

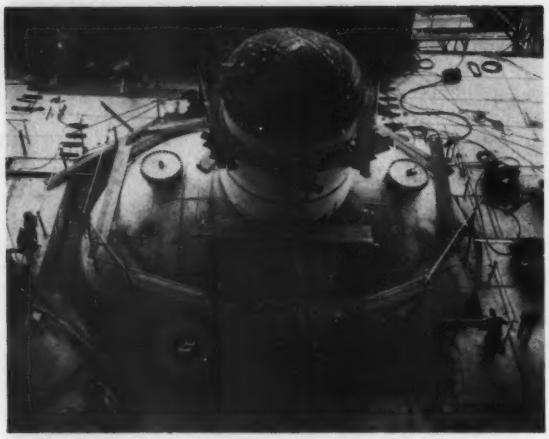
Intermediate Cooling System. Fresh water at 95°F is provided for cooling the primary circulating pumps, the primary shield tank, and the gaseous waste collection system, in addition to the air conditioning system for the containment. This fresh water is itself cooled by sea water, through two heat exchangers.

Emergency Cooling System. Should all electrical power supplies fail except the 300 kW emergency diesel, a portion of the primary coolant can be pumped through a heat exchanger directly cooled by salt water, to remove decay heat from the core. Salt water in this system can also cool the containment cooling system and the emergency pump.

Shielding and Containment

One of the most important features of any nuclear ship is the method of containment of activity—in all foreseeable circumstances—and the attenuation of radiation to a negligible amount, and *Savannah's* arrangements are by no means the least interesting feature of her design.

The containment vessel was designed to withstand the pressure produced in case of a rupture of the primary circuit and the consequent "flashing" of the primary and secondary water, this being set at 186 p.s.i. It was also designed to resist the stresses set up by some 500 tons of shielding with the dynamic stresses set up by a 30° roll. Details of the



(Left) The containment vessel during hull construction.
(Below) Fixing polythene shielding slabs.

design of the vessel are given in an appendix; it will here suffice to say that the containment is a hemispherical-ended cylinder, 35 ft diameter and 50 ft 6 in. long with plate thickness varying from 1.24 in. to 3.75 in., encompassing a volume of 41,300 ft^3, and weighing 275 tons. A dome, or cupola, with a 13 ft opening is provided on the top of the vessel, with a 42 in. access opening forward and aft of the cupola. Two standard manholes are provided on the underside.

The vessel is supported on six saddle-shaped supports welded to the ship's structure. In order to allow for expansion, the vessel is bolted only at the after end of the cylindrical portion, with 24 high-tensile bolts 2¼ in. diameter. This locates the vessel and resists rotation, and the remaining supports are fitted with lubrite bearings and pads. Other supports at 90° in way of the collision mats, and chocks between B and C deck, maintain the vessel in position, even under a 90° heel.

The containment is normally maintained gastight, and air-conditioning maintains the inside temperature at about 120°F. Ventilation only takes place when the reactor is shut down, when the exhaust is continuously monitored.

There are, in all, 76 penetrations of the shell, varying from 1 in.-12 in. diameter, in addition to the main access openings and manholes. Nozzles are used for piping smaller than 2 in.; above this size a reinforcing doubler is used with the nozzle. For cables epoxy resin seals are used.

Shielding has been divided into primary shielding, inside the containment, around the reactor itself, and secondary shielding, around the containment.

Primary shielding consists of an annular tank surrounding the reactor and 17 ft high giving a layer of water 2 ft 9 in. radial depth; around this there is lead, varying from 2-4 in. thick, bonded to the tank. This is sufficient to limit the dose rate within the containment to 200 millirem/h within 30 minutes of the shutdown, so that limited access for maintenance work is possible.

Secondary shielding, intended to reduce radiation levels to negligible amounts throughout the ship, is considerably more elaborate, and consists of nearly 2,000 tons of lead, polythene, concrete and water. The complex geometry of the secondary shield was tackled by a number of programmes written for an IBM-650 digital computer.

The lower part of the vessel is enclosed in a concrete wall, high-density concrete being used where space limitations prevented an adequate thickness of ordinary concrete; a total of 508 tons of ordinary and 552 tons of high-density concrete being used. The wall is extended forward of the containment to form a compartment colloquially known as the "doghouse," in which are located low-activity items of the plant, such as the purification demineralizers and components of the waste collection, buffer seal and gaseous waste systems.

The upper portion of the shield consists of lead and polythene. The thickness of the lead varies from 2½-6 in., the slabs being bolted on to the containment vessel and the interstices caulked with lead rope.

Below the containment, a series of fresh water shield tanks in the double bottom of the ship follow the line of the concrete and compensate for radiation scatter under the shielding.

PROPULSION PLANT AND AUXILIARIES

The main propulsion machinery space is situated aft of the reactor space, between bulkheads 6 and 7, which gives a space 55 ft long, 78 ft wide, and 32 ft high from the tank top to C deck.

Main Turbine

The main turbine is a De Laval two-cylinder unit, rated at 20,000 s.h.p. normal (22,000 max.) geared to give a normal propeller speed of 107 r.p.m. (110 max.). A moisture separator is located in the crossover line between the high- and low-pressure sections. The astern turbine is an integral part of the L.P. section and delivers about 80% of full load (ahead) torque at 50% of normal revolutions. The condenser is designed for 28.5 in. vacuum at a seawater temperature of 75°F. Seawater is provided by scoops during normal operation; for manœuvring and astern working, the main circulating pump provides cooling.

The reduction gear also makes provision for engaging a 750 h.p. electric motor for emergency operation.

Electrical Plant

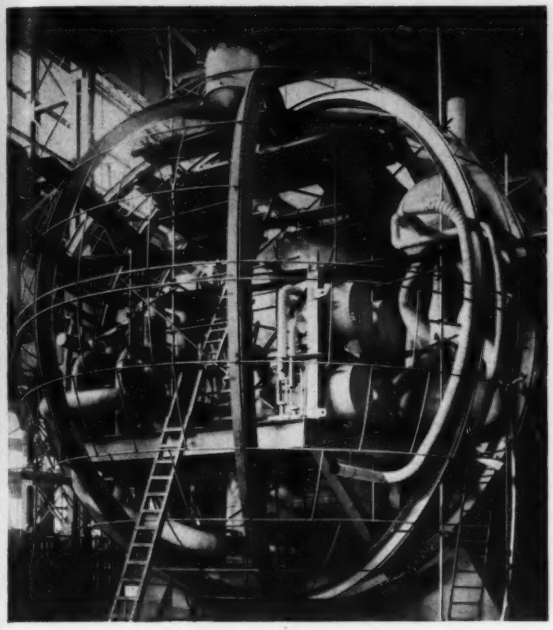
Two 1,500 kW turbo-generators normally supply the reactor auxiliary power, as well as the "hotel" load of the ship, and ship's auxiliaries, such as cargo winches, etc.

There are also two 750 kW diesel generators which can be used for "take-home" power in emergency, and will operate auxiliaries during reactor start-up. They are also available for decay heat removal during reactor shut-down and, in case of a scram shut-down, are automatically run up and synchronized.

"Last-ditch" protection is available in the form of a 300 kW diesel generator, located on the navigating bridge deck which, in case of failure of all other supplies, will operate emergency lighting, the low-speed windings of the reactor coolant pumps, and the emergency cooling system.

D.C. supplies for the reactor are provided by two motor generator sets, fed from the main switchboard, and connected to a 125 V d.c. bus, on which a battery floats, being always maintained fully charged and ready in case of power failure. The d.c. bus also provides closely regulated a.c. supplies, through two 25 kW d.c./a.c. motor generator sets, for instrumentation and certain vital loads.

(Below) The reactor containment mock-up. (Right) The reactor control simulator at Lynchburg.



Radiation Monitoring

There are three channels devoted to health physics. Channels 1 and 2 each allow indication and, if desired, recording of the levels of six selected areas within the ship. Channel 3 covers three access detectors giving background levels at points of entry into the containment or the "doghouse," to indicate, after shut-down, whether entry is permissible or not.

Five channels are devoted to the functioning of the plant. Channel 4 has five detectors for possible leaks of activity from the primary system to the intermediate cooling water; Channels 5 and 6, with one detector each, search for leakage into the secondary system. Channel 7 detects the presence of fission products in the primary coolant due to fuel cladding failures, while Channel 8 monitors the ion exchangers on the water purification system to see if the exchange resins require renewing.

Eight channels are devoted to the monitoring of the waste disposal plant; four measuring particulate activity, and four gaseous activity, in the stack.

In addition to the various measurements linked to the control room, there are many items of portable equipment for on-the-spot measurements. There are, in addition, more than 400 locations throughout the ship where film tags are exposed, to keep a cumulative record of radiation.

Safety

The complete safety assessment of the ship has not yet been published, and it is only possible to review the problems in general terms. From an analysis made of marine accidents taking into account the size of ship and the damage resulting, it is believed that the *Savannah* could withstand collision with all but about 1% of the world's merchant shipping; the precautions taken in the matter of containment have already been described. In the same way, the immensely strong construction of the hull and bottom are expected to render her immune from damage due to grounding. As explained in the appendix on page 459, the ultimate danger of crushing of the containment and release of fission products on sinking is guarded against by manhole covers designed to give way at a certain depth, and reclose on equalization of pressure.

A team of experts is to visit each port which it is proposed to visit, several weeks in advance of the ship, so that all local and national requirements regarding safety in port can be settled in advance.

Mock-up, Simulator, and Training Programme

Some idea of the thoroughness with which the *Savannah* project has been tackled may be gathered from an examination of the reactor mock-up. This is a full-size model of the nuclear power plant, located at the New Jersey shipyard of the New York Shipbuilding Corporation. It is complete in every detail, with large components, such as the reactor and heat exchangers, built in light-gauge sheet metal, and cardboard, plastics, etc., to simulate the smaller details and piping; and ladders and gratings in aluminium.

There are several reasons for this equipment. First, and most important, is the training of the crew, who will be able to move about in this unit in exactly the same way as in the real plant. Maintenance routines can be worked out and tested; many of the components are designed to this end. A primary pump or gate valve can be removed from the loop and, by means of the lifting gear provided, can be taken out through a 42 in. manway; the operation of plugging a defective heat exchanger tube or replacing a pressurizer heating element can be practised in detail.

In addition, the mock-up has been of considerable value during construction; it has been possible to check designs for clearances and interferences prior to the installation of gear aboard the ship.

Experience in reactor control is provided by a reactor simulator designed and built by the Westinghouse Electric Corporation and located at the U.S. Maritime Reactor School at Lynchburg, Virginia. The cost of the simulator was $180,230.

Training commenced in September, 1958, for 16 licensed engineer officers in reactor theory, engineering and operation. Twenty additional students from outside the States Marine Corporation also took this 15-month course. An additional accelerated course was also taken by a number of officer candidates, and foreign technicians sponsored by the AEC. Other courses were also held for deck officers.

Only minor constructional features of the *Savannah* remain to be completed; the full core has been assembled and tested over the first six months of this year and fuel loading in the vessel is due to begin in December. Initial steam production is expected in January and full power operation in March, 1961.

Following the successful conclusion of dockside power trials, she will undergo sea trials and, after working-up, will probably undertake limited commercial operation for about two years. She will then, in all probability, be chartered to a private operator to be used in normal commercial service. . . . Good luck to all who sail in her!

SAVANNAH

Control Rod Operation

THE control-rod system for the reactor of the *Savannah* is more easily understood if it is regarded as three separate sub-systems:—

1. A control-rod drive assembly, with 21 electro-mechanical drive mechanisms, and their associated switchgear and auxiliaries.
2. A hydraulic system to provide scram action and load balancing during normal operation.
3. An electrical control and instrumentation system for switching, interlocking, monitoring and servo control of both the above.

It will readily be seen that for a marine reactor, with its absence of fixed verticals and its liability to sudden movements in almost any direction, gravity cannot be relied upon for scram insertion of the rods. Also, since the control-rod drives must, of necessity, penetrate the top plate of the reactor, there is a considerable "piston" force, due to the internal pressure acting on the control rod. The hydraulic system not only completely overcomes this force when scram action is required, it furnishes a downward force at all times which partly balances the thrust during normal operation, relieving the load on the electromechanical operating mechanism, and minimizing wear, as well as reducing the load on the motors.

The mechanical and the hydraulic systems are completely separate up to the point of their action on the control-rod drive line in such a way that any defect of the mechanical system cannot in any way impair the operation of the hydraulic scram system.

The entire set of mechanisms actuating the control rods is mounted vertically on the upper head of the pressure vessel, each drive mechanism actuating its control rod by means of a drive line shaft, which penetrates the closure head through nozzles. Each nozzle is equipped with a "buffer seal" to prevent reactor water from leaking outward along the shaft.

Control-rod Drive Assembly

Each of the 21 drive mechanisms is attached at its lower end to a thimble platform, as shown in Fig. 1. The thimble platform is mounted on the upper end of a buffer seal assembly which, in turn, is mounted on a control-rod nozzle extending upwards from the top of the reactor vessel. The control drive mechanism is approximately 17 ft long from the top of the motor to the nozzle with a spacing between nozzles of approximately 9.7 in. between centres.

The thimble platform contains a split shroud which surrounds the actuating shaft, and prevents chips, dirt or other material from coming in contact with the actuating shaft at this point and, in the event of a failure of the buffer seal, prevents contamination of adjacent drive mechanisms. The thimble platform also contains the lower mechanical stop which prevents the drive mechanism from putting the control rod and lower drive line into compression. Access is provided through the split shroud to uncouple the lower drive line from the control rod prior to removing the reactor head.

The drive mechanism is shown schematically in Fig. 2. It consists of two motor-driven lead screws powered by a two-phase servomotor through a reduction gear. The output shaft of the gearbox is directly coupled to one lead screw, which is connected by a chain and sprocket drive to the other. The lead screws move the carriage against which the actuating shaft is loaded by reactor pressure.

The position of the carriage is indicated continuously by a selsyn position indicator mounted on the drive mechanism which transmits to a selsyn receiver in the control room. Limit switches cut off drive motor power at the upper and lower limits of normal control rod travel. The rod stroke and velocity are adjustable up to a maximum stroke of 72 in. and between 5 and 15 in./min respectively.

A "drive pin," through the actuating shaft, is held against the underside of the drive carriage by the upward force of reactor pressure on the drive line. To reduce the loading on the drive nuts and lead screws, this force is opposed by a downward force applied by a controlled pressure in the upper cylinder or actuator. This "balance" pressure varies with reactor pressure over the range of 1,500 p.s.i. to 2,000 p.s.i., so as to maintain a net upward force of approximately 500 lb at all times. The rotation of the motor-driven screws causes the carriage to move; the actuating shaft then follows the carriage

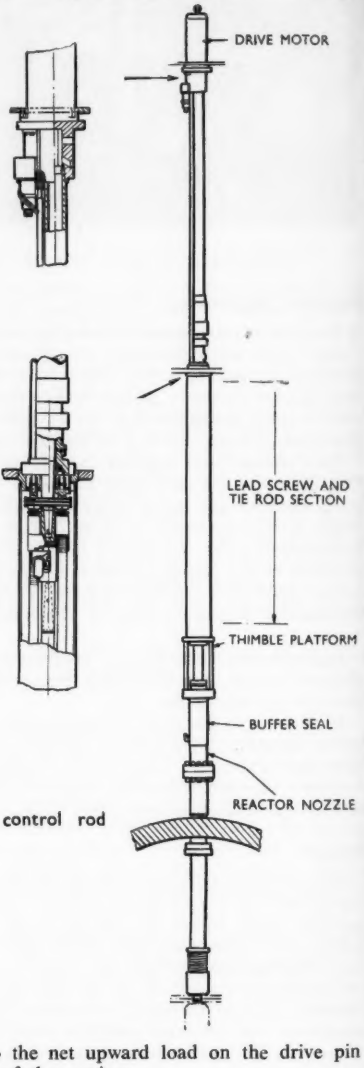


Fig. 1.—Diagram of control rod drive.

vertical travel due to the net upward load on the drive pin against the underside of the carriage.

Hydraulic System

A simplified diagram of the scram portion of the hydraulic system is shown in Fig. 3. It provides four basic functions:

TYPE:

PURPOSE:

OWNER:

MAIN CONTRA

STATUS:

OUTPUT:

FUEL:

CORE:

FUEL ELEMEN

CONVERS FACTO

BURN-UP:

CONTRO

COOLANT

PRESSURE VESSEL:

HEAT EXCHA

The World's Reactors

No. 28 N.S. *SAVANNAH*

TYPE: Thermal, heterogeneous (PWR).

PURPOSE: Marine Propulsion Experiment.

OWNER: U.S. Maritime Administration.

MAIN CONTRACTOR: The Babcock and Wilcox Company.

STATUS: Critical 1960.

OUTPUT: Design, 74 MWt; normal 63.5 MWt; m.c.r., 69 MWt.

FUEL:
Enriched uranium oxide (4.4% av.).
Initial loading 312.4 kg U^{235}

CORE:
Cruciform prism, composed of 32 channels 8.5 in. (215.8 mm) square, arranged in rows of 4-6-6-6-6-4, and baffled for 3-pass cooling.
Equivalent diameter: 62.06 in. (1,376 mm).
Active height: 66 in. (1,676 mm).
Overall height: 90.24 in. (2,291 mm).
Volume fractions:
- Water 0.5655.
- Control rods 0.0405.
- Helium gap 0.0052
- Fuel 0.2647.
- Stainless steel 0.1421.

Excess reactivity for:

Temperature (including power coefficient)	3.2%.
Xe and Sm	2.0%.
Power Doppler	1.3%.
Burn-up and isotope build-up	4.7%.

Total excess reactivity 11.2% $\Delta k/k$.
Neutron flux (thermal, average): 7.2×10^{13} n/cm², sec.

FUEL ELEMENTS:
Rod bundle type, with 164 rods, in four bundles of 41 rods at 0.663 in. (16.9 mm) centres.
Diameter of oxide pellets: 0.4255 in. (10.8 mm).
Thickness of stainless steel cladding: 0.035 in. (0.89 mm).
Outside diameter of rod: 0.5 in. (12.7 mm).

CONVERSION FACTOR: 0.4.

BURN-UP: 7,352 MWd/t (average).

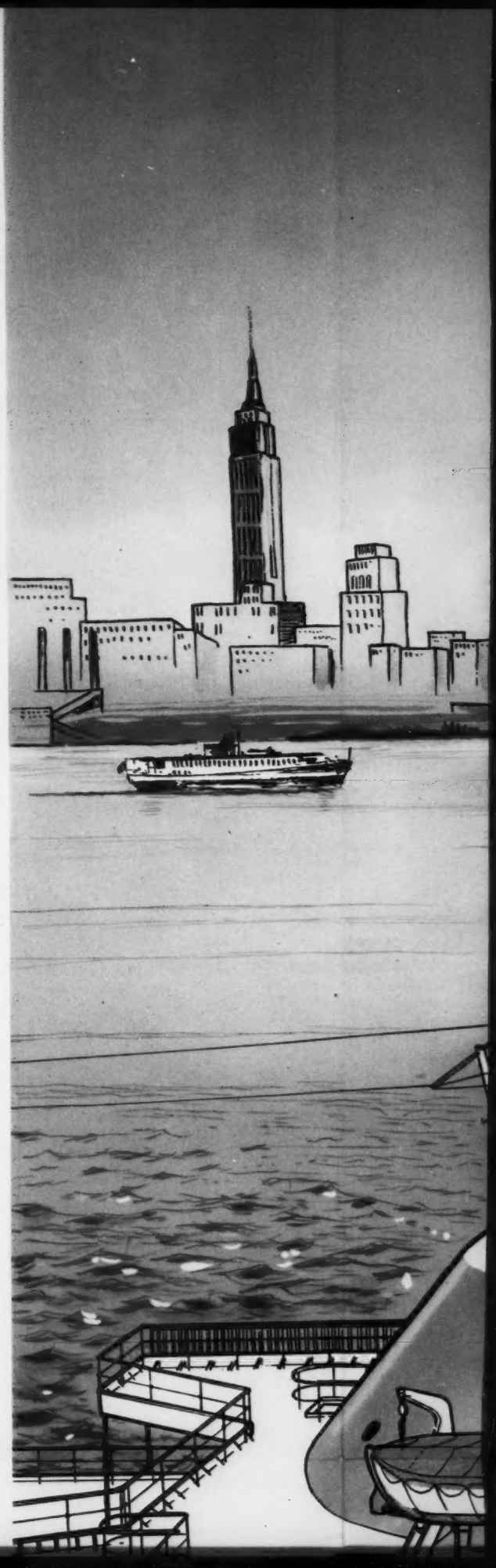
CONTROL:
Cruciform section boron stainless steel, clad in stainless steel, 8 in. (203 mm) square × 0.375 in. (9.52 mm) thick.
No. of rods: 21.
Effective length: 66 in. (1,676 mm).
Total worth of control rods: 14–18% $\Delta k/k$.

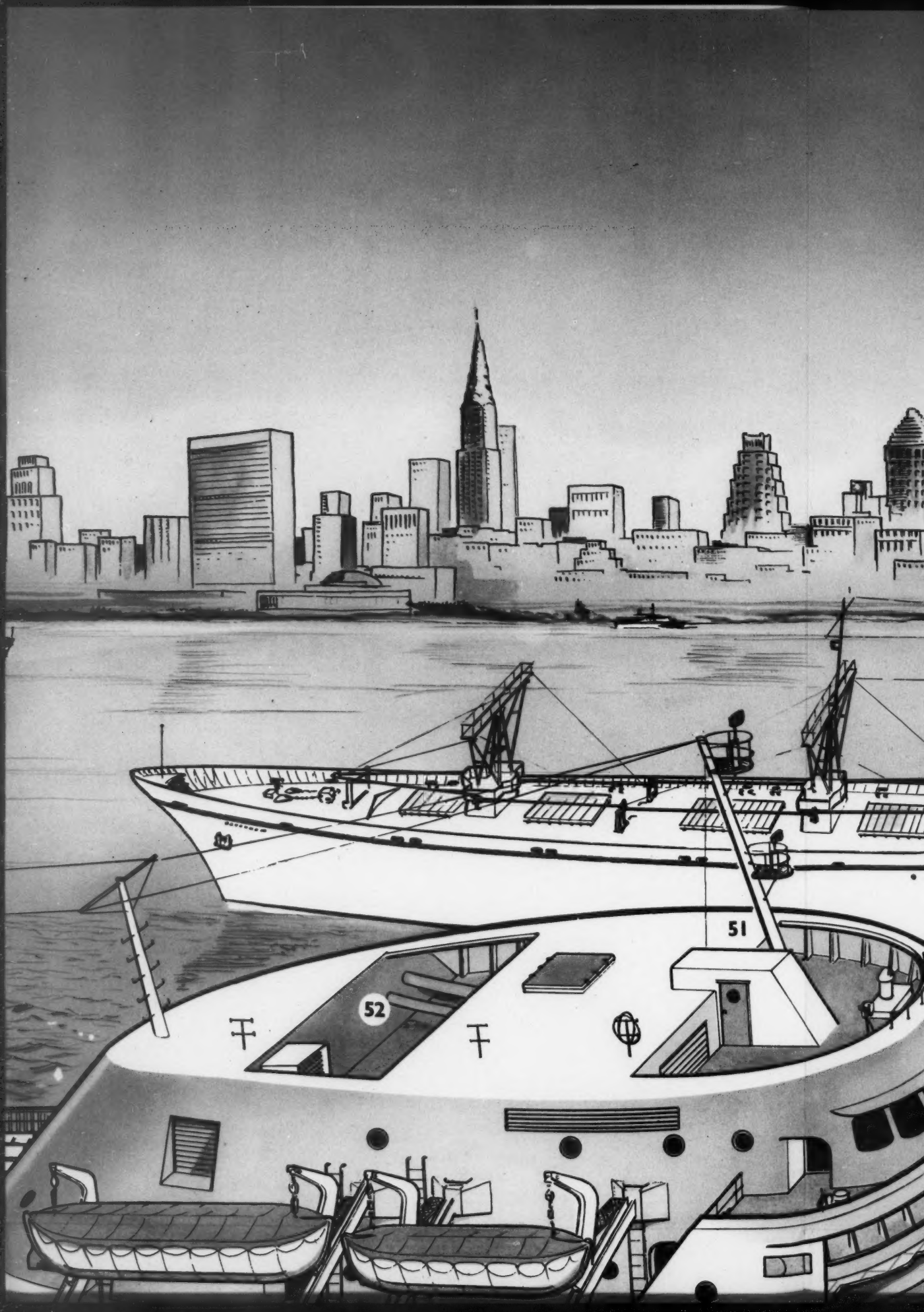
COOLANT:
Pressurized water, in three vertical passes.
Inlet temperature: 495.6°F (257.5°C).
Outlet temperature: 520.4°F (271.3°C).
Hot channel: 541°F (282.7°C).
Max. fuel element temperatures:
- Fuel: 3,794°F (2,090°C).
- Cladding: 623°F (328.3°C).

Heat transfer area: 3,778 ft² (350.8 m²).
Coolant pressure 1,750 p.s.i.a. (123.04 kg/cm² abs.).
Mass flow: 8×10^6 lb/h ($3{,}625 \times 10^6$ kg/h).
Average velocity in core:
- Outer pass: 9.29 ft/sec (2.83 m/sec).
- Inner pass: 8.4 ft/sec (2.56 m/sec).

PRESSURE VESSEL:
Carbon steel, clad internally with stainless steel.
Height: 26 ft 6 in. (8.077 m).
Inside diameter: 8 ft 2 in. (2.489 m).
Thickness: 6 in. (152.4 mm).
Design pressure: 2,000 p.s.i.a. (140.614 kg/cm² abs.).

HEAT EXCHANGER:
Two primary loops, each with two circulating pumps and one heat exchanger of the U-shell, U-tube type, having 800 ¾-in. tubes on the primary side. Secondary (shell) side is connected by 13 risers and 8 downcomers to separate steam drum with cyclone separators and steam scrubbers.
Design pressures:
- Tube side: 2,000 p.s.i. (140.614 kg/cm²).
- Shell side. 800 p.s.i. (56.246 kg/cm²).

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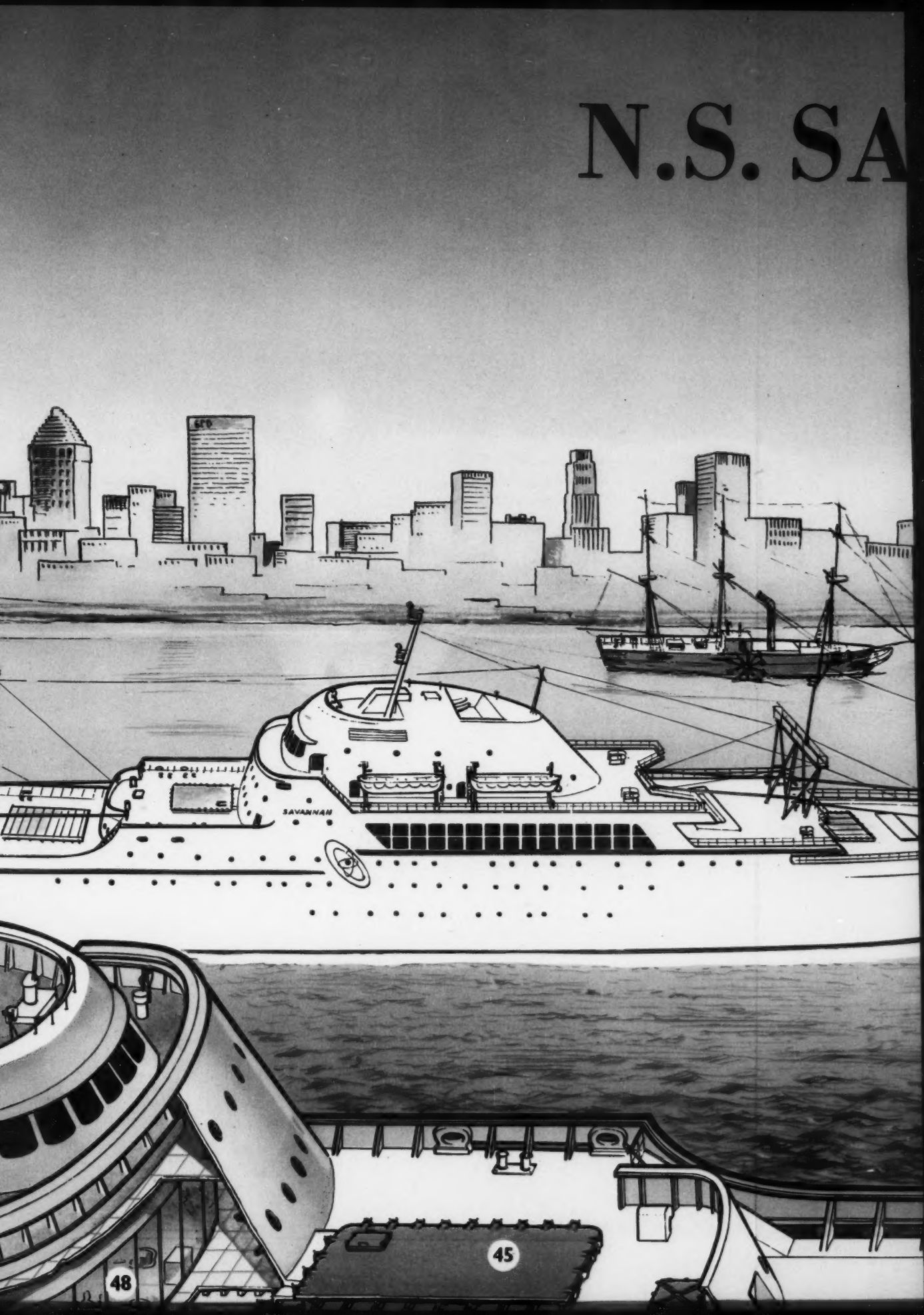
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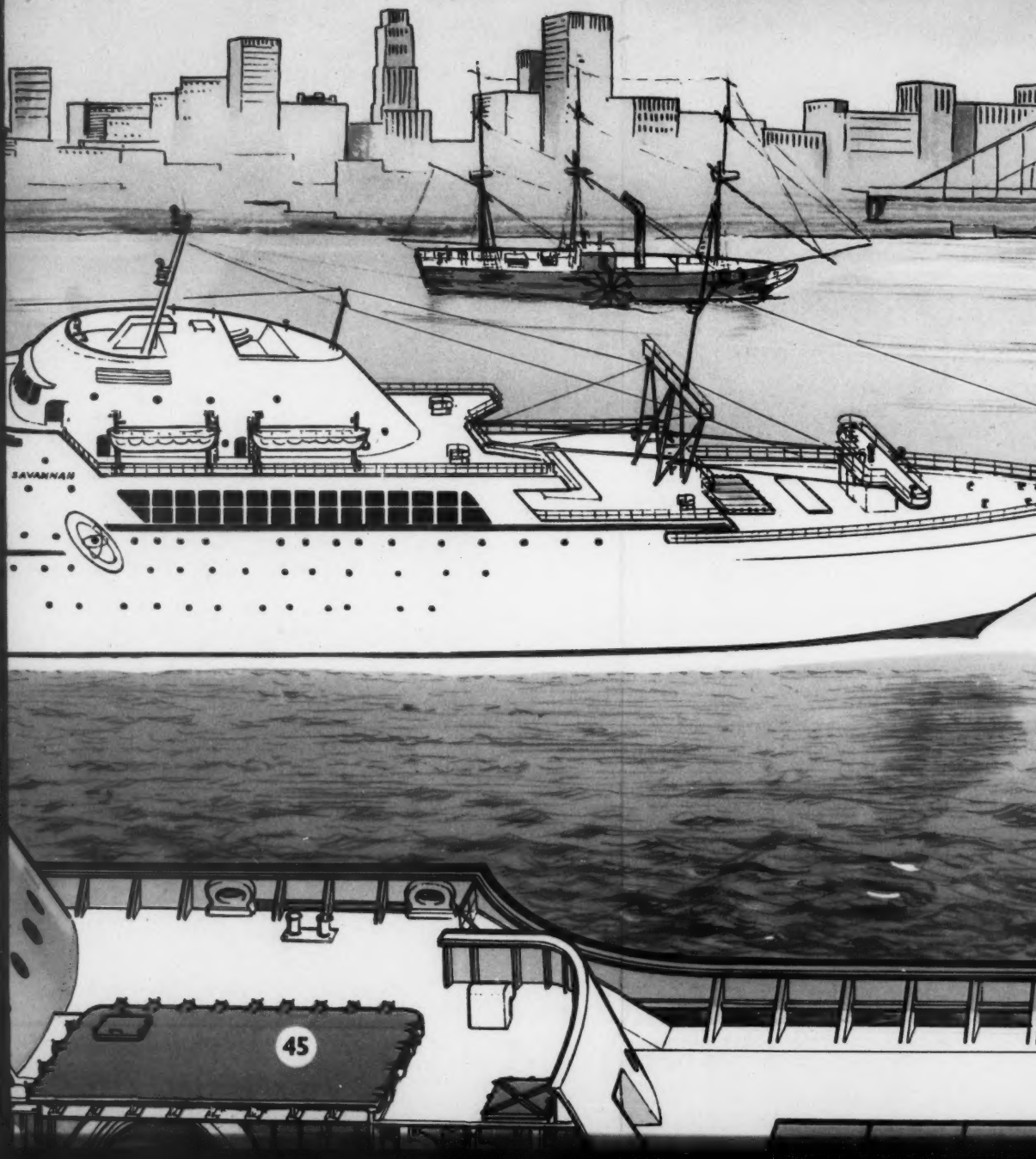


ANNAH

KEY

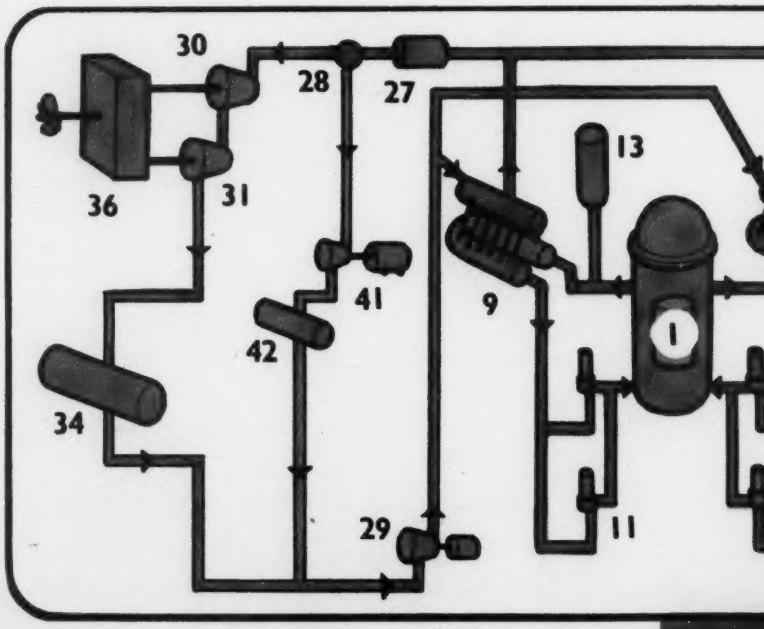
1. Core
2. Control rods
3. Control rod mechanism
4. Reactor shell
5. Lead covered neutron shield tank
6. Primary inlet to reactor
7. Primary outlet
8. Gate valve
9. Heat exchangers (2)
10. Steam drums
11. Primary pumps
12. Surge line to pressurizer
13. Pressurizer
14. Condensing tank
15. Drain tank
16. Let-down coolers
17. Cooling ducts
18. Steel containment vessel
19. Lead shielding
20. Polyethylene shielding
21. Vessel support members
22. Water treatment plant
23. Concrete shielding
24. Steel and redwood laminated collision mat
25. Anti-roll brackets
26. Secondary steam outlet to turbines
27. Main steam separator
28. Throttle valve
29. Main feed pumps and secondary return to heat exchangers
30. H.P. turbine
31. L.P. turbine
32. Steam separator
33. Take-home motor
34. Main condenser
35. Main condenser circulating pump
36. Reduction gears
37. Thrust bearing housing
38. Coupling
39. Propeller shaft steady bearing
40. Propeller shaft
41. Turbo-generators sets (2)
42. Auxiliary condenser

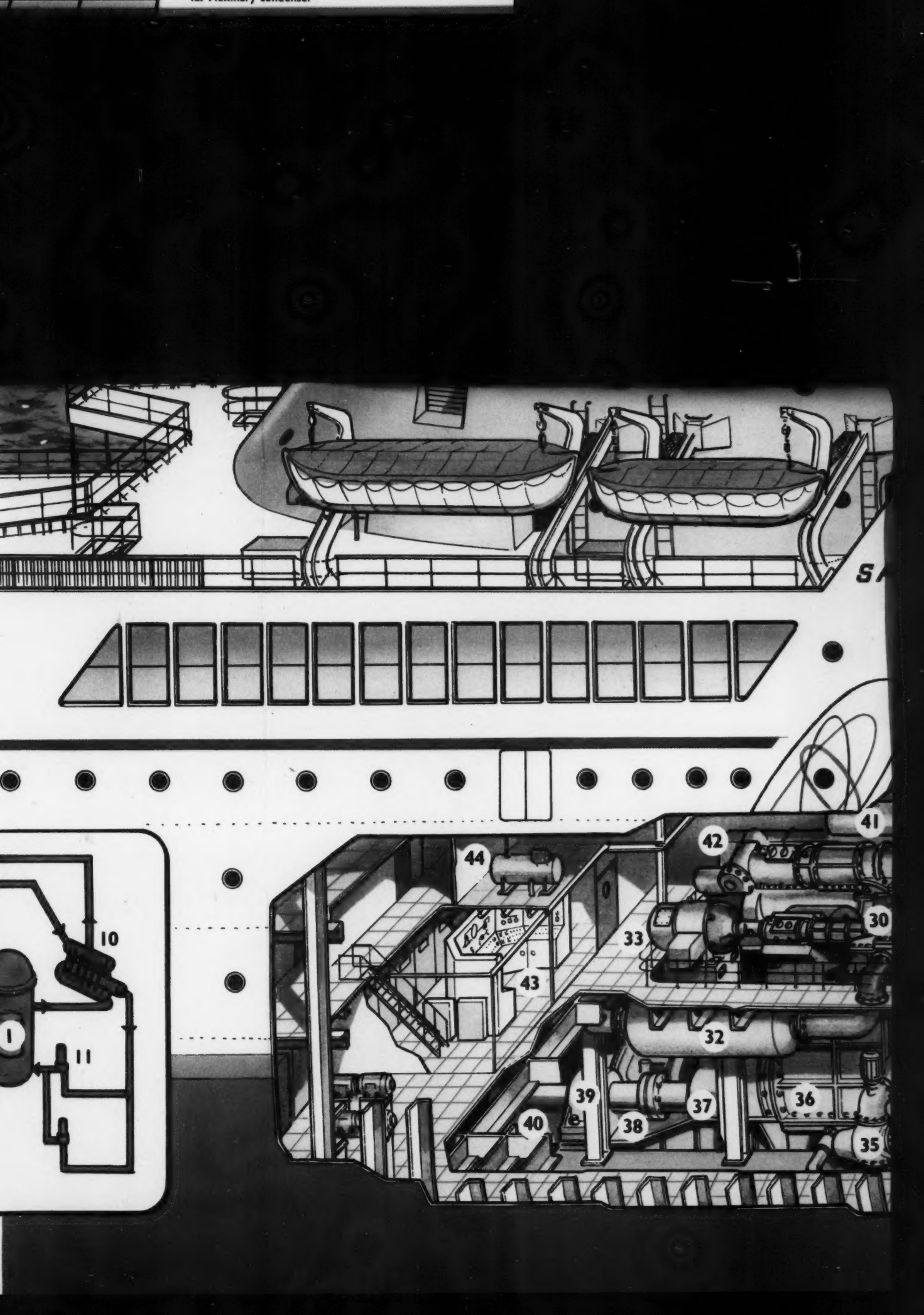
Inside diameter: 8 ft 2 in. (2.489 m).
Thickness: 6 in. (152.4 mm).
Design pressure: 2,000 p.s.i.a. (140.614 kg/cm² abs.).

HEAT EXCHANGER: Two primary loops, each with two circulating pumps and one heat exchanger of the U-shell, U-tube type, having 800 ¾-in. tubes on the primary side. Secondary (shell) side is connected by 13 risers and 8 downcomers to separate steam drum with cyclone separators and steam scrubbers.
Design pressures:
Tube side: 2,000 p.s.i. (140.614 kg/cm²).
Shell side. 800 p.s.i. (56.246 kg/cm²).
Design temperature: 650°F (343.3°C).
Steam conditions at normal load: 490 p.s.i.a. (34.45 kg/cm² abs.) dry saturated (0.25% moisture).
Feed temperature: 347°F (172.6°C).
Total steam generation at normal load:
242,000 lb/h (110,000 kg/h).
Total primary pumping power (4 pumps): 920 kW.

SHIELDING: Primary shielding (inside containment): Steel tank 17 ft (5.182 m) high, surrounding reactor with an annular thickness of water of 33 in. (0.838 m), surrounded by 2-4-in. thickness of lead. Secondary shielding (outside containment) is a composite arrangement of polythene up to 6 in. thick, lead, ordinary and barytes concrete, and water tanks, and collision mat of alternate layers of redwood and steel.

CONTAINMENT: Steel cylinder 35 ft (10.668 m) diameter, with hemispherical ends, overall length 50 ft (15.24 m). Thickness varies from 2½ in. (60.3 mm) to 4 in. (101.6 mm).

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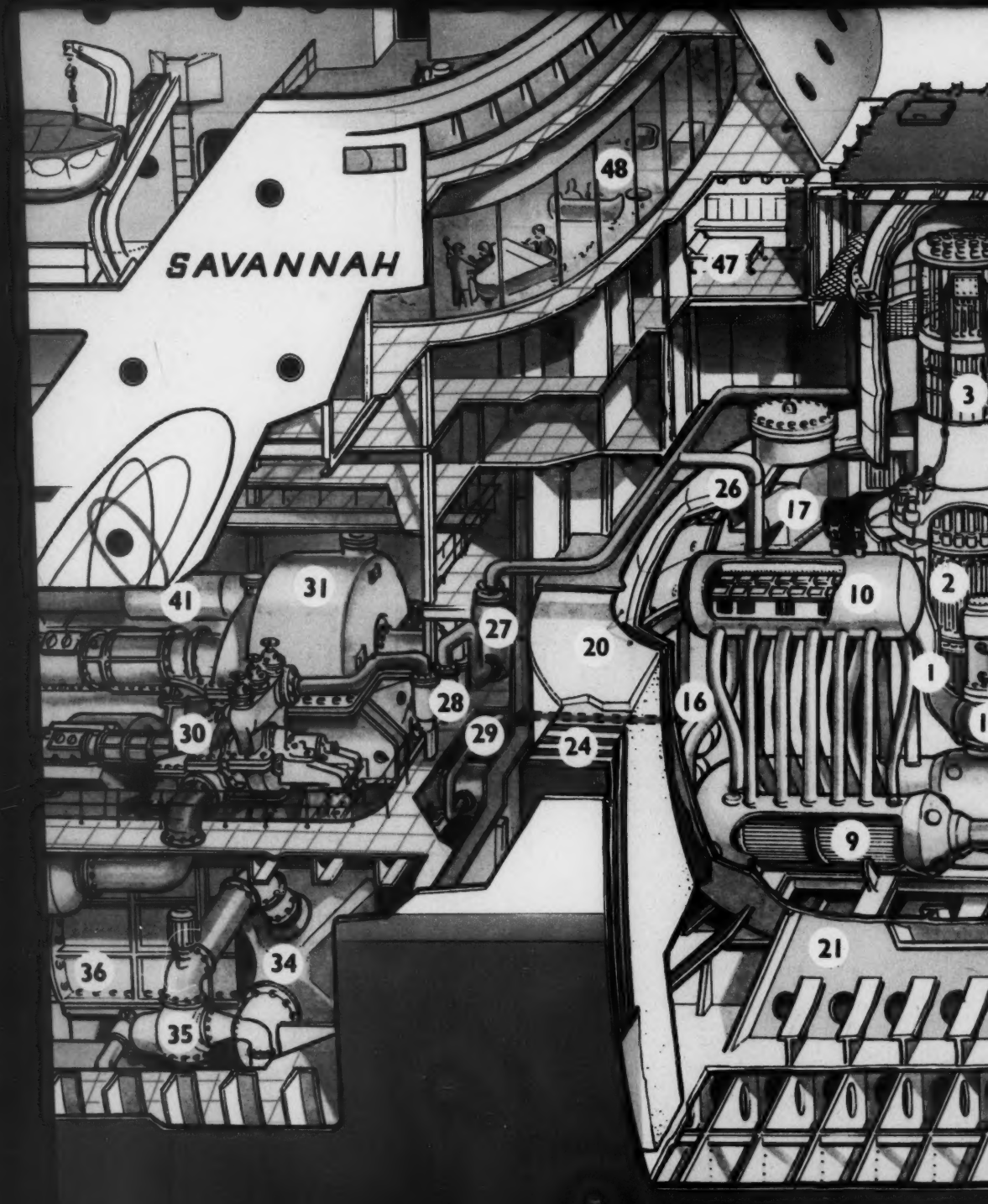
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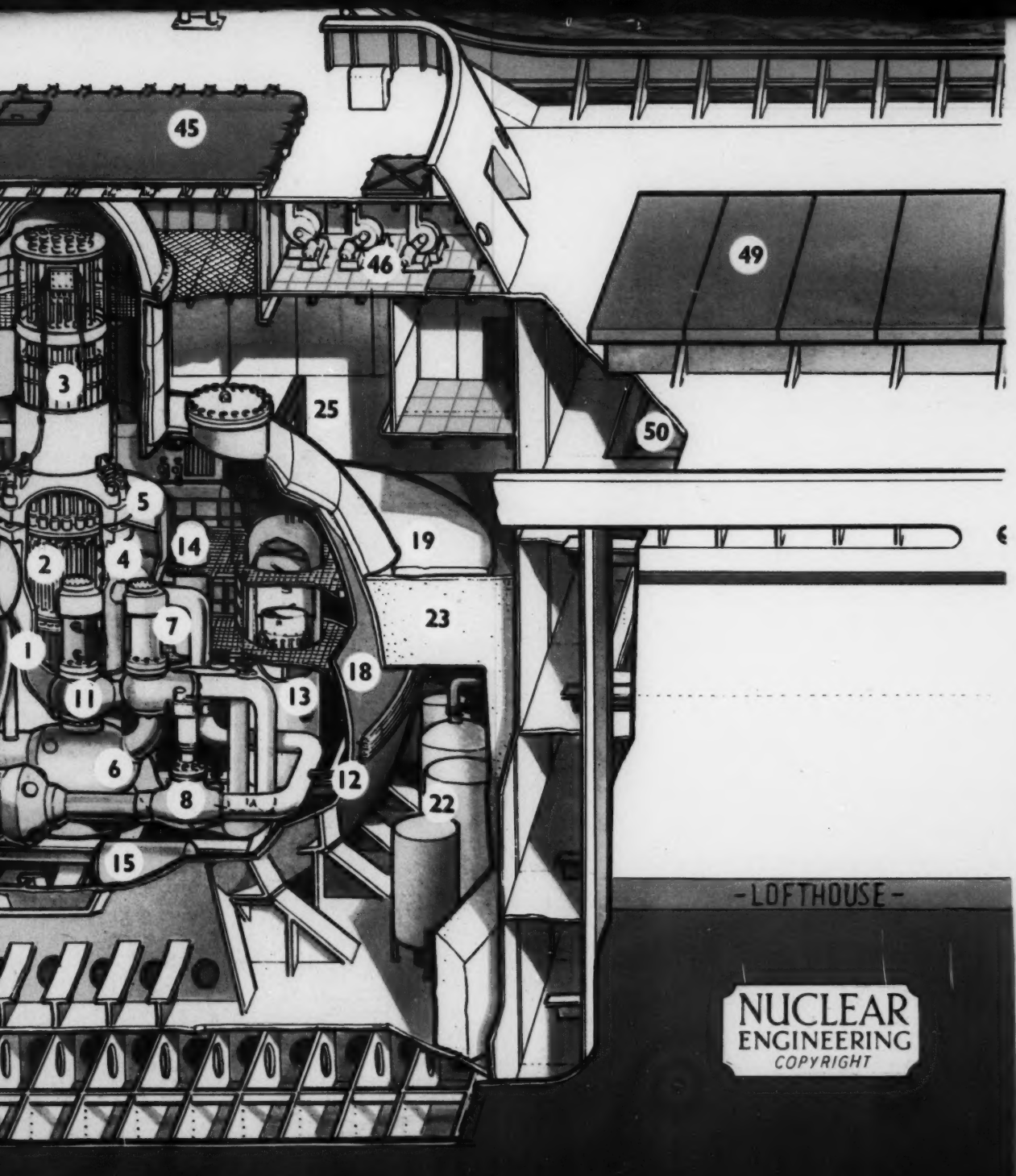

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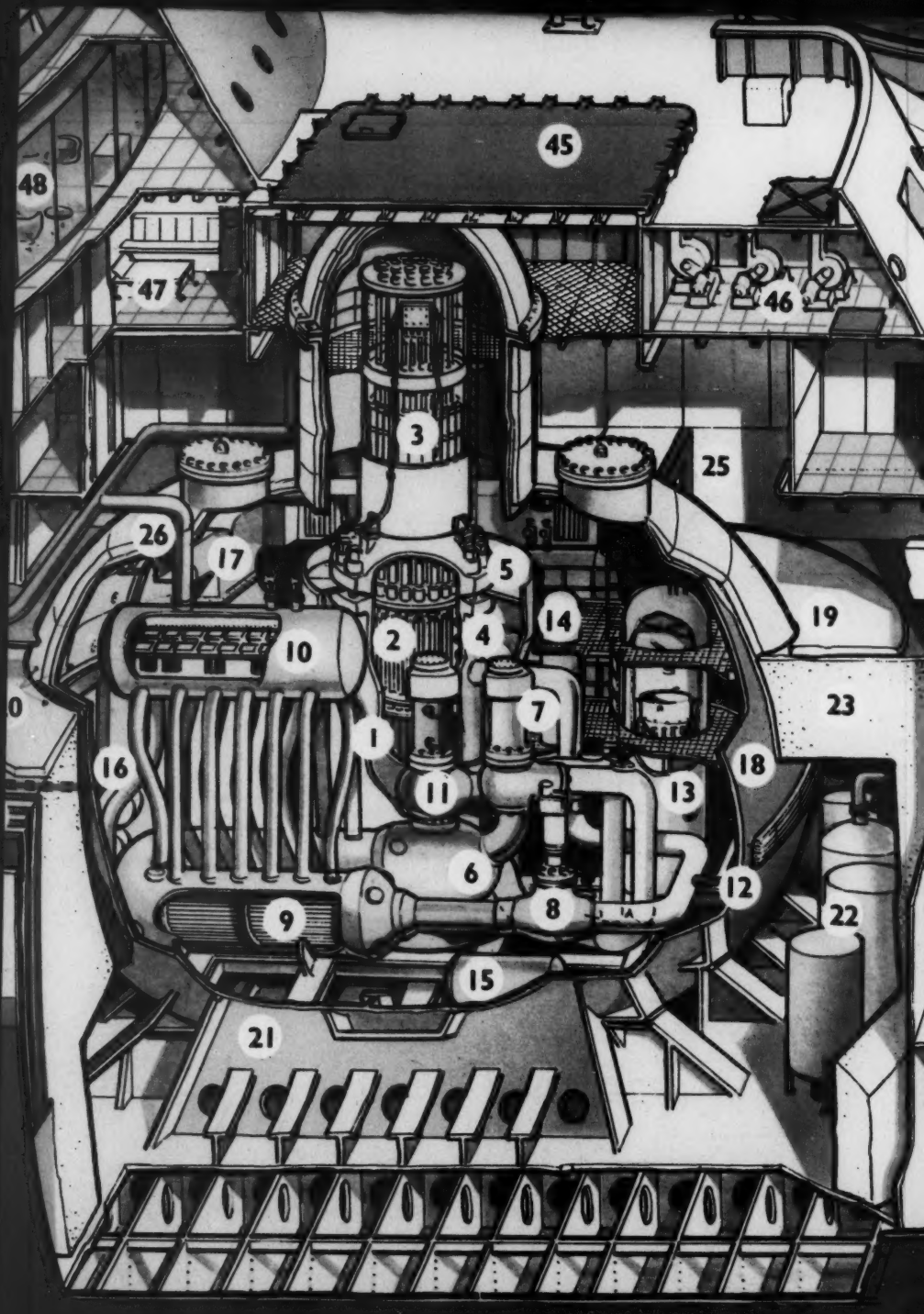
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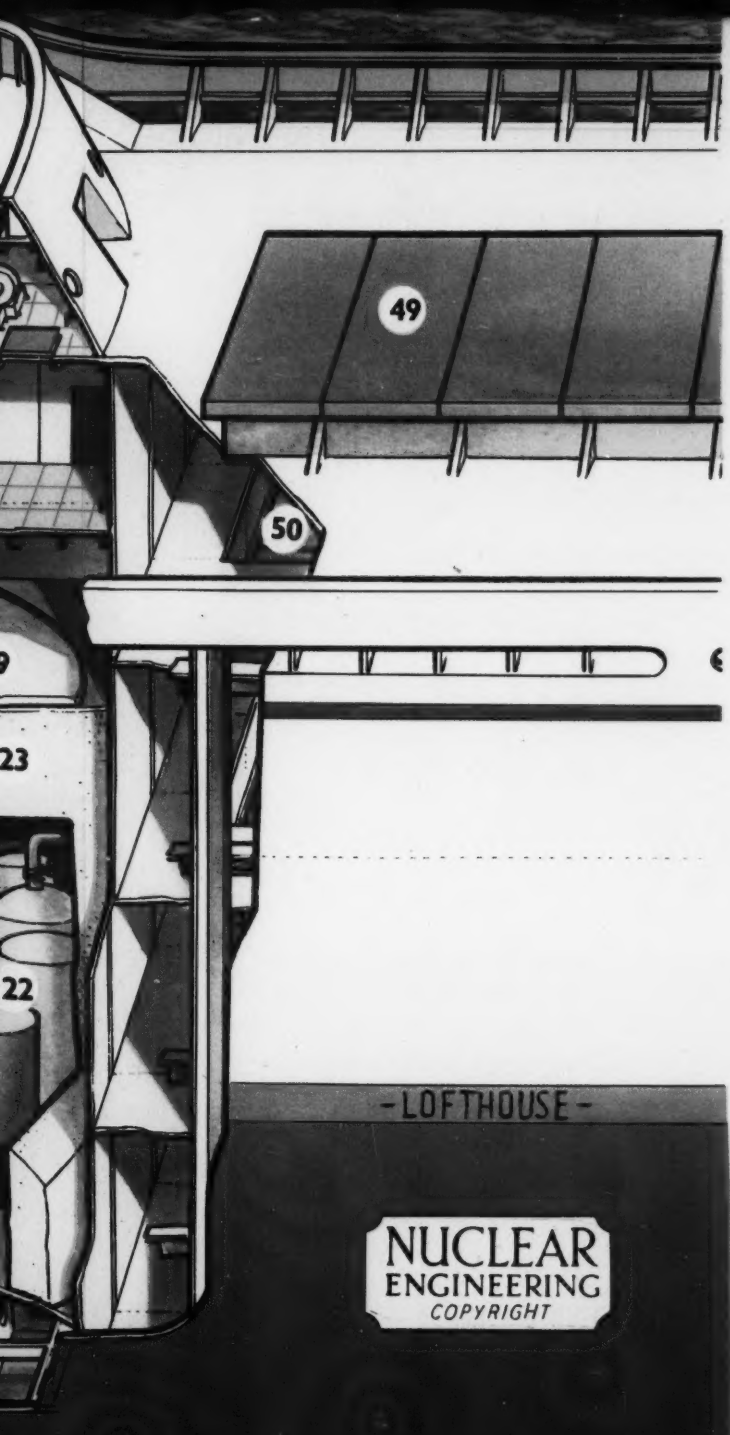

36. Reduction gears
37. Thrust bearing housing
38. Coupling
39. Propeller shaft steady bearing
40. Propeller shaft
41. Turbo-generators sets (2)
42. Auxiliary condenser
43. Control centre
44. Water heater and tank
45. Reactor hatch
46. Fan room
47. Sick bay
48. Main lounge
49. Cargo hatch
50. Cargo hold
51. Waste gas disposal channel
52. Air conditioning exhaust

DETAILS OF SHIP

Length between perpendiculars	545 ft
Length overall	595 ft 6 in.
Beam	78 ft
Draught, fully loaded	29 ft 6 in.
Displacement, fully loaded	22,000 tons
Displacement, light	12,000 tons
Deadweight	9,900 tons
Cargo capacity	9,400 tons
Horsepower, (normal)	20,000 s.h.p.
Horsepower, (maximum)	22,000 s.h.p.
Speed	21 knots
Passengers	60
Crew	110
Cargo capacity	746,200 ft³
Fuel radius (estimated)	300,000 naut. miles

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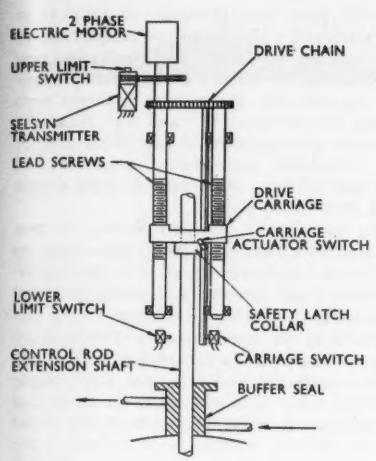
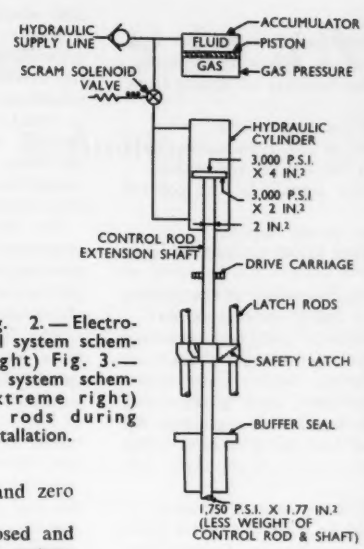
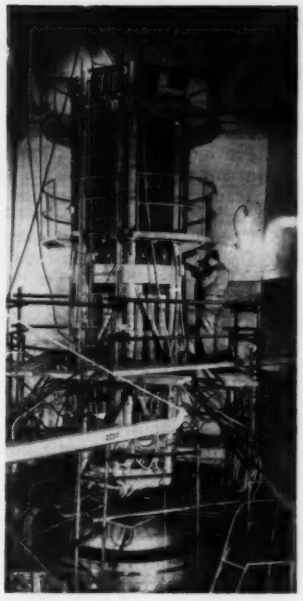






(Left) Fig. 2. — Electro-mechanical system schematic. (Right) Fig. 3.— Hydraulic system schematic. (Extreme right) Control rods during installation.

scram action, velocity control, pressure balancing, and zero reactor pressure lifting force.

The hydraulic system scram valve is normally closed and opens only on receipt of a signal from the reactor safety system. When it opens, high pressure oil flows from the accumulator to the top of the hydraulic cylinder, accelerating the rod up to a constant velocity limited by a calibrated orifice in the external hydraulic circuit. As the piston in the hydraulic cylinder approaches the end of its normal scram stroke, it is decelerated and brought to rest with a controlled deceleration force. The scram system is designed to scram each rod from a fully withdrawn position to two-thirds insertion in 0.8 sec after receipt of a scram signal.

The hydraulic cylinder provides sufficient force at the end of a scram stroke to hold the control element in the scram position indefinitely without any other assistance or electric power. Nevertheless, if the hydraulic pressure should decrease to a level where reactor pressure could cause the control rod to be forced out of the reactor, two latch arms carried on the actuating shaft engage two latch rods which run the full length of the screw-driven section of the drive mechanism. One latch alone is sufficient to hold the control rod. This latching operation prevents the actuating shaft from rising regardless of the attitude of the ship, until the latches are released by the carriage being driven down against the drive pin by the motor. The carriage itself then becomes an additional factor preventing a control rod being driven out of the core by reactor pressure.

Hydraulic Power Supply

Three separate hydraulic power supply units are installed in the power supply room located between frames 110 and 114, B deck level, on the starboard side of the containment cupola. Any one of these can be assigned to operate the system, by means of automatic or manual switching, with the other two available on standby or reserve. Each power supply unit consists of an electrically driven high-pressure piston pump, a main relief valve, a cooler, a reservoir, a low oil level alarm, an oil temperature indicator, a suction strainer, an in-line filter, an isolating check valve, and a manual isolation valve.

The three hydraulic power supplies are connected to a common high-pressure header which is monitored by four pressure switches. The operational unit supplies the flow required to the high pressure header while the second unit is maintained in a standby capacity. The third unit is held in reserve and is manually switched into operation upon failure of either of the other two.

The condition of failure of either the operational or standby unit is annunciated by pressure switches so that the defective unit can be isolated manually and the reserve unit placed in service. Interflow between the three high-pressure power supply units is prevented by check valves, providing maximum reliability of operation and safety of the primary source of scram energy.

Buffer Seals

Each control rod drive line passes through a buffer seal mounted on the nozzle through which it penetrates the reactor head, as shown in Fig. 1. All 21 seals are identical. Essentially, the buffer seal is a device by which purified water from the outlet of the buffer charge pumps is imposed upon the shaft seal in such a way that a portion flows inward at all times to the reactor and the remainder flows outward through the upper seal rings to a low pressure plenum whence it is returned to the buffer surge tank. A vertical section of a buffer seal is shown schematically in Fig. 4. The high-pressure incoming flow, maintained automatically 50 p.s.i. above reactor pressure, flows inward through five seal rings at a rate of about 0.5 gal/min per seal. The portion of the inflow passing outward along the drive line shaft is also limited to about 0.5 gal/min by 18 seal rings. The pressure in the plenum chamber is approximately 40 p.s.i.a., determined by buffer surge tank pressure. This chamber is sealed off by a relatively soft "atmospheric" seal through which the normal leakage does not exceed a few drops per hour.

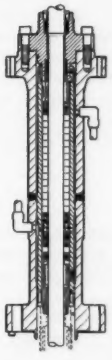


Fig. 4.—Section of buffer seal.

When the control rod is fully inserted, a shoulder on the drive line shaft penetrates the atmospheric seal sufficiently to seal it off completely even against full reactor pressure. By this means, if the charge pumps should fail, the primary system pressure can be held indefinitely merely by driving in all control rods.

The entire cartridge of 23 seal rings plus atmospheric seals is replaceable, and is also separated from the buffer seal housing by a thermal barrier. The entire assembly has been tested to withstand the same pressure as the reactor vessel itself, 3,000 p.s.i.

Special prototype tests on the buffer seal assembly alone were carried out by the manufacturer to the extent of 60,000 ft of travel plus 500 scram strokes. After such testing, considerably in excess of estimated lifetime travel, the only evidence of leakage from the atmospheric seal consisted of a few drops of water adhering to the shaft during withdrawal. Inward and outward leakage through the multiple rings did not increase measurably during these tests.

Controls and Instrumentation

The electrical controls and instrumentation associated with the control rod drive system perform the following functions:

1. Operate the control rod drive mechanisms on signals from the reactor control system.
2. Provide closed loop control of the servo drives.
3. Provide an interconnection panel to select the servo drives at any time in accordance with the control rod worths.
4. Transmit rod position and limit signals to the control console.
5. Control power to the hydraulic power supplies.
6. Operate and interlock each scram valve, as well as other valves.
7. Monitor the critical and noncritical portions of the control rod drive as required for safety and proper operation.

The electrical control system consists of eight servo-controls governing the motors on the central ring of eight rod drives, plus 13 manual motor controls on the others. The servo-controlled rods form two groups of four, each group being driven synchronously. Differences in load within a given group do not affect the synchronization of rod group motion. The system also permits manual operation of any of the eight rods in the two rod groups.

Each of the eight servo-controlled loops consists of an input selsyn transmitter which is positioned by a shaft angular input signal derived in one of the group demand units. Output from the selsyn is compared to a feedback signal on the drive mechanism and fed to a high gain magnetic amplifier. The output from the magnetic amplifier drives the 300 W two-phase servo drive motor. Tachometer and position feedback signals from the rod drive mechanisms are compared to the control rod group input signal in the magnetic amplifiers.

A magnetic switching device in the magnetic amplifiers compares desired input signal to the actual output signal and derives an error signal. If the error signal exceeds a preset value, the failure monitor operates a relay to return the drive mechanism to manual control.

The manual motor controls provide for constant-speed operation of the 12 rods, either individually, or in three groups of four, by action of the control room operator only. The core centre rod, "X," is not assigned to a group, and is manually operated.

The instrumentation of the control rod drives consists of position indication and limit switches. The position indication components are 21 selsyn position indicator transmitters mounted on the drive mechanisms and 21 individual plus five group position indicator receivers located in the control room.

The position indication system provides a continuous indication of the position of each carriage. The limit switches cut off power to the drive motors at the extremes of normal mechanism travel. They are also used to calibrate the position indicator system.

Equipment Arrangement

The components of the control rod drive system are located in five areas; within the containment, the main control room, and in the three power supply rooms on "B" deck, port and starboard of the containment.

The main control room houses the group demand units, the selsyn position indicator receivers and the control console.

The three power supply rooms are as follow:

1. Containing the control cubicle housing the servo amplifiers and failure monitors, the patchboard, and the pump controls used for control of the control rod drive system, all located at B deck level, port side, aft of frame 116.
2. Containing the three hydraulic power supply units and associated accessories, and located at B deck level, starboard side, between frames 110 and 114.
3. Containing power equipment such as the Scott-T transformers, and located just forward of the control cubicle, room No. 1.

Prototype Mechanism Tests

The prototype control rod drive mechanism was first fabricated and subjected to a complete "life" test. This prototype mechanism closely resembled the production mechanisms in that standard production parts and methods of fabrication were used wherever significant. The prototype test served to uncover any minor defects in the design, and to demonstrate the ability of the mechanism to meet rigorous performance conditions. For this purpose, the programme included tests in which the drive line was inclined 15° and 30° from the vertical, with and without misalignment of simulated reactor internals, and under simulated atmospheric conditions of temperature, humidity and salinity. As expected, some design improvements resulted from these tests.

The prototype testing was divided into two phases; the first included 30% of the life test of the drive mechanism only; in the second, the satisfactory performance of the mechanism and its supporting equipment was demonstrated under simulated shipboard conditions during an accelerated life test.

The first phase, consisting of approximately 14,000 ft of travel and 137 scrams, was completed in November, 1958. Tests at 15° inclination were conducted under normal atmospheric conditions whereas vertical tests (80% of total) were conducted at 150°F and 90% humidity. A saline environment was maintained by open tanks of sodium chloride solutions within the test cubicle.

In the second phase, the entire drive line was tested including the seal shaft, control rod and buffer seal. The control rod and simulated reactor internals were contained in an autoclave capable of maintaining the reactor pressure and temperature. Provisions were included for simulating misalignment of drive line components. A main circulating loop and a buffer seal loop provided flow conditions comparable to those expected in the reactor. Typical variations in reactor water purity were also used in the test. A chronological summary of the prototype tests is presented in Table 1.

Table 1. Prototype Control Rod Drive System Tests

Tests on Drive Mechanism		Travel, ft	Scrams
Phase I:	Vertical, no misalignment of vessel internals	10,000	181
Phase II:	Vertical, no misalignment	14,000	150
	Inclined 30°, no misalignment	500	29
	Vertical, with misalignment	9,000	73
	Inclined 15°, with misalignment (97 hr continuous operation)	8,000	50
Final Test:	Vertical, no misalignment	5,000	50
	Total	46,500	533
Separate Tests on Components			
Buffer Seal (at Stein Seal Company):	1st type	32,000	350
	2nd type	60,000	500
Control Rod and Follower		21,000	200

Drive Mechanism and Hydraulic System Tests

1. Drive motor—speed and electrical load tests under extreme conditions of friction and balance pressure.
2. Zero balance pressure—check run-in capability at 2,000 p.s.i. simulated reactor pressure.
3. Lead screw lubrication—inspection.
4. Latch tests—with two latches, perform three tests at different positions with 2,000 p.s.i. simulated reactor pressure. Repeat for each latch individually.
5. Drive position tests.
6. Breakaway tests.
7. Stability tests.
8. Ramp tests—determine maximum and minimum ramp rates.

Production Mechanism Tests

The test and inspection programme for each of the production mechanisms is listed in Table 2. During these tests, a total of 12 hours' operation under various operating conditions was carried out on each mechanism.

Table 2. Tests on Production Mechanisms

Component Inspection and Performance Tests

1. Hydraulic cylinder—friction, hydrostatic and leakage tests.
2. Hydraulic valve panel—hydrostatic, internal leakage and operational tests.
3. Hydraulic power supplies—leakage, noise, vibration, pump delivery and hydrostatic tests.
4. Hydraulic accumulators—leakage and hydrostatic tests.
5. Magnetic amplifier—sensitivity, gain, and transient response.
6. Thimble platforms—100% dimensional inspection.
7. Support structure—dimensional and alignment checks.
8. Motor and gear box—torque, efficiency, back-lash, slippage and break-loose torques, and leakage.
9. Group demand unit—gain adjustment, sensitivity stability and calibration.

SAVANNAH

Refuelling

By J. D. McDANIELS, Jr.,
(Remote Handling Group Supervisor, The Babcock and Wilcox Company Atomic Energy Department)

WHILE the method selected for the refuelling of the N.S. *Savannah* is similar in concept to that adopted for EBWR, N.S. *Seawolf* (sodium cooled), and a host of PWR type nuclear submarines, considerable interest has been created by the introduction of several unique components and design approaches. Notably in this category are the reactor vessel head stud tensioning operations and the design approaches permitting operations to be safely and easily carried out throughout a liberal range of ship motions and attitude deviations.

General

The basic refuelling task consists of the replacement of the material in Table 1.

Table 1. Material Handled During Refuelling

Item	Size	Weight	Dose Rate	Heat Output
32 Fuel Elements	8¾ in. sq. × 76½ in.	760 lb	10^6 r/h	66,000 B.t.u./h
21 Control Rods	8 in. sq. × 144 in.	230 lb	10^5 r/h	—
2 Neutron Sources	0.35 in. dia. × 24.5 in.	¾ lb	8 rem/h	—

Note: Dose rates are approximate levels 5 days after shutdown, with the exception of the neutron source.

In order to gain access to these core components the containment and reactor vessel heads must be removed, followed by the removal of two large core hold-down and flow baffle components. The upper flow baffle, weighing 17,432 lb and emitting 7 r/h, is removed first, followed by the removal of the upper grid assembly weighing 5,510 lb and emitting 3,500 r/h gamma.

To remove the fuel and control rods, a large rotating shield manipulator plug is placed on the reactor vessel head and the components are withdrawn, one at a time, into a transfer cask that is resting on, and located by, the manipulator plug. The cask, with a fuel element or control rod within, is then removed from the containment vessel and transported by crane to a special floating service vessel, where the element or rod is placed into storage for later shipment to the reprocessing centre. The new control rods and fresh fuel elements are then inserted into the vessel with two 100 curie sources and all retaining and closure components replaced to permit start up tests to proceed.

Motion Parameters

To ensure reliability and operational ease, the refuelling components were designed to the following liberal motion parameters:

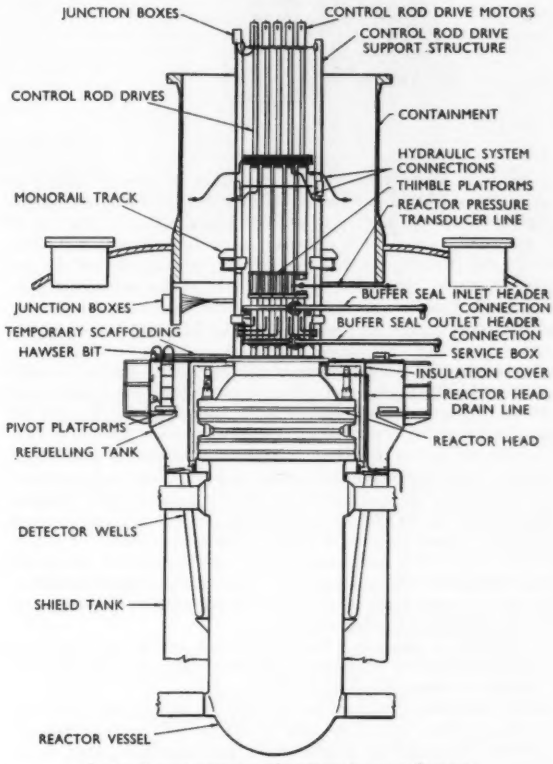
1. Maximum heave of ship: One inch total vertical displacement with a minimum heave period of seven seconds.
2. Maximum pitch of ship: ½° amplitude pitch superimposed on 1½° steady trim with a minimum pitch period of seven seconds.
3. Maximum roll of ship: 1° amplitude roll superimposed on 2° steady list with a minimum roll period of 15 seconds.
4. Maximum deviation from true vertical: 3.6°.
5. Maximum drift from pier: One foot.

NOTE: All periods refer to a complete cycle and all amplitudes refer to maximum displacement from centre.

Preparation for Head Removal

After the main reactor hatch, located on the promenade deck, is set aside, the containment vessel head, with shielding attached, is removed and stored. A prefabricated "clean house" is next placed over the hatch opening to provide protection from inclement weather and aid in ventilation control. The structure will also be the control centre for all activities as facilities will be provided for clothes change, personnel monitoring, equipment accountability, and operations direction.

With the clean house in place, ventilation equipment in operation, and entry monitoring completed, the operators secure the control rod drives by breaking a shaft connection just above each control rod drive buffer seal, rotating the shaft 45° to unlock the control rod from the drive line, and reconnecting the shafts so that they may be withdrawn to the full out position, leaving the control rods in the core.



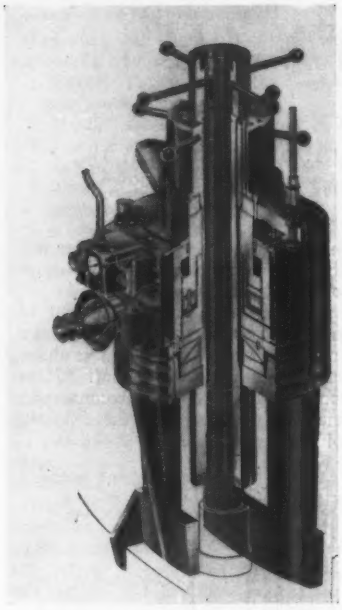
General arrangement of reactor in containment.

All control rod drive actuator and buffer seal electrical and hydraulic lines are disconnected and 18 insulation cover segments are removed from around the reactor vessel head followed by replacement of two BF_3 proportional counters located on opposite sides of the reactor vessel and the installation of a seal between the refuelling extension tank and the reactor vessel flange. The ship attitude indicator is installed within the refuelling tank, and head-removal operations may proceed.

Reactor Vessel Head Removal

The reactor vessel head is secured with 48 studs of 5 in. diameter, and sealed with two metal O-rings backed by a seal-welded membrane.

In the past, tightening or detensioning procedures for large studs have used heaters to lengthen the studs and permit rotation of the nut. Due to the lengthy heating and cooling time requirements (approximately 48 hours) experiments were performed on



(Left) Cutaway view of the Diamond 600-ton stud tensioner.

(Right) The stud tensioner undergoing full-scale tests.

3½-in. studs, using large impact wrenches to power tighten or loosen the nuts. While appreciable time reductions were realized, one out of four stud-nut assemblies seized either at the nut or in the vessel. Later improvements, such as polishing and silver plating the threads, reduced the number of seizures, but did not eliminate them. In addition, the noise level was extremely high, resulting in fatigue and operational errors, and steel slivers were occasionally ejected as the result of the hammer blows on the nuts.

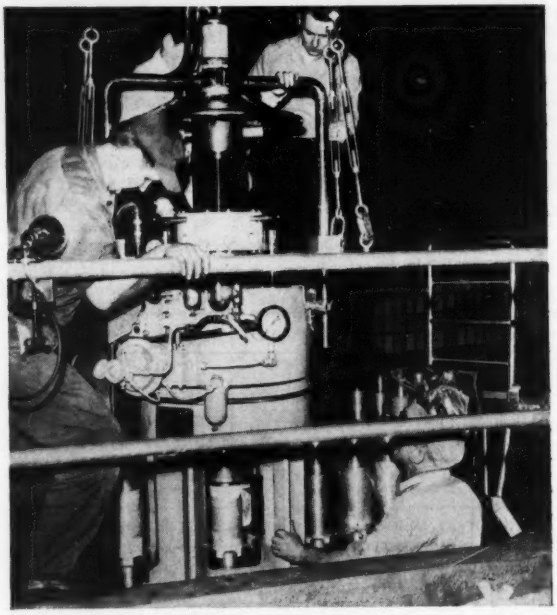
To reduce turn-around time of the *Savannah* and yet provide a safe and reliable device to remove or tighten the stud nuts, a 600-ton capacity stud tensioner was conceived by B. and W. and developed by a subsidiary, Diamond Power Specialty Corporation, of Lancaster, Ohio. This device stretches the stud by means of a hydraulic cylinder which applies a load to a drawbar that is screwed on to the stud portion extending past the nut. The reaction of the cylinder is taken by the vessel flange through a cylinder support casting straddling the row of studs. The tensioner moves from one stud to the next by means of an air motor powered trolley suspended from a monorail attached to the reactor vessel head control rod drive structure. The tensioner has the following unique features:

1. Unit is suspended by four cables acting as a multiple parallelogram, to maintain angular alignment with the stud regardless of the ship's attitude.
2. Tensioner lifts itself by the introduction of air above the power piston, which in turn is affixed to the four support cables. Since the air cannot drive the supported piston down, the tensioner cylinder housing rises to permit relocation on the next stud.
3. During cutting or welding of the reactor vessel seal membrane, six studs remain to hold the head down against the internal flow sealing springs. The tensioner drawbar may be locked to a stud to permit its retraction into the reactor vessel head to permit cutting or welding of the area behind it. Following cutting or welding, the stud is power reinstalled into the vessel and tensioned all without moving the tensioner unit.
4. While the tensioner operates on oil pressures up to 10,000 p.s.i., the unit operates off plant compressed air as a pump-pressure intensifier and oil reservoir are built in.

Continuing with the head-removal operations, all but six studs are relaxed and then removed by means of a stud handling trolley. The studs are transported to storage racks by means of a deck or dockside hoist and a seal weld-cutting machine is attached to the reactor vessel head. The cutter is tracked by grooves machined on the head and is held in place and driven by silent chains encircling the head. An air motor driven end mill removes the seal weld in one pass as the six remaining studs are relaxed, retracted, installed and retensioned as the cutter passes by.

Following clean-up, the remaining studs and weld cutter are removed and head jack-screws are affixed to the head to permit it to be raised past close-tolerance alignment keys, regardless of the ship's attitude.

A novel six-legged sling, designed to snub out rotary motion, is used to transport the 135,000-lb head assembly complete with

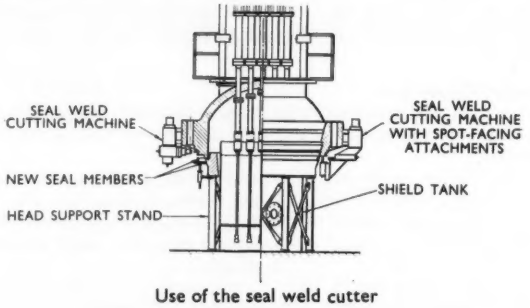


drive actuators and their support structure from the containment to a dockside head maintenance stand where a new seal weld membrane and O-ring are installed.

Removal of Core Hold-down Internals

To protect the vessel flange seal surfaces after removal of the head, a protective ring is lowered into place, guided by the same guide shafts used in the removal of the head.

The core hold-down internals to be removed for access to the fuel consist of two assemblies. The first to be removed is the



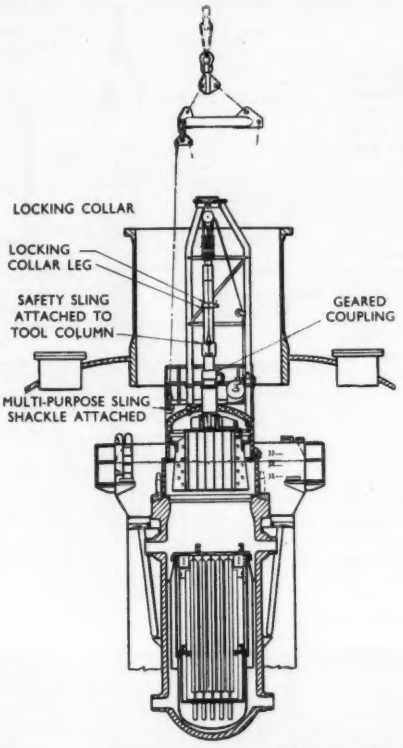
Use of the seal weld cutter

upper flow baffle, which will have an approximate dose rate of 7 r/h. The upper grid plate is removed next and due to its close proximity to the core will have an approximate dose rate of 3,500 r/h.

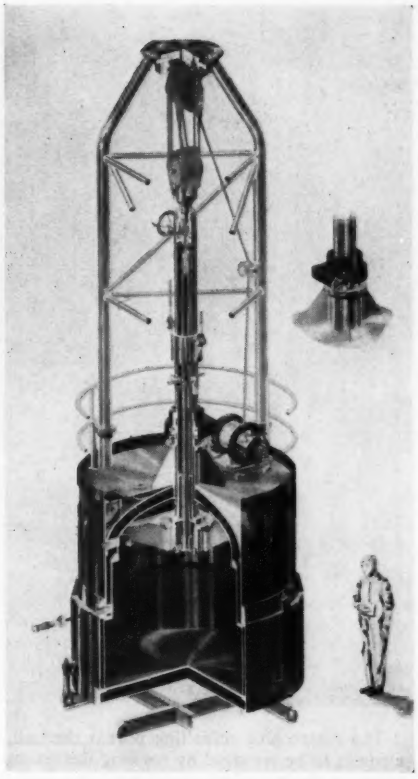
Since the flow baffle is more than twice the height of the hotter grid plate, it was possible to design a telescoping internals handling transfer cask to permit the same shield cask to handle both units without a resulting weight penalty. The shielding was designed to handle the smaller but hotter grid plate with the transfer cask telescoped. By extending the two concentric shells

of the cylindrical section of the cask the larger upper flow baffle may be handled with sufficient shielding.

The internals cask is equipped with an integral hoisting system for withdrawal of the internals from the reactor vessel and into the shielded cask. This feature, combined with a well-supported piston-like tool column, permits the handling of the internals in and out of the reactor vessel, without need for compensating for the ship's motion and attitude. To meet test specifications, the internals have been removed at a 3.6° attitude on a full scale mock-up without difficulty.

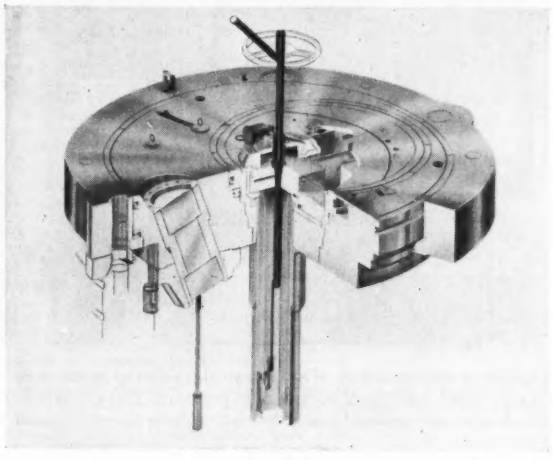


(Right) Cutaway view of telescopic internals handling cask.



(Left) Internals handling cask removing upper flow baffle.

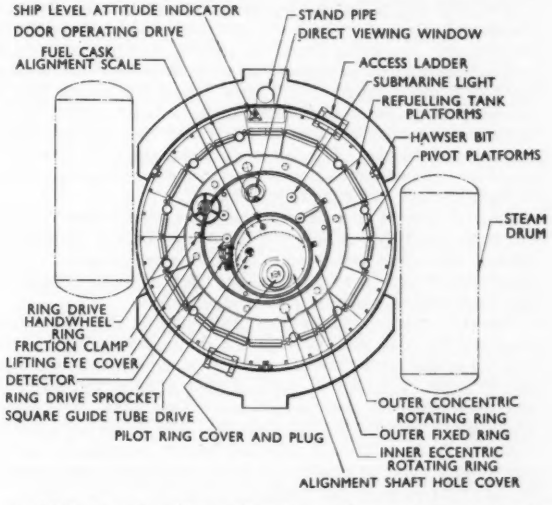
After the internal components have been withdrawn into the cask aboard the ship, personnel vacate the containment to avoid the radiation emitted from the underside of the cask as it is lifted off the reactor vessel. The cask is transported to, and placed upon, the door section located on the promenade deck. After the door is attached, the cask is transported to the upper flow baffle storage container and the upper flow baffle is transferred from the cask to the container. The cask is then telescoped and returned to the ship for the upper grid assembly. This assembly is removed in a like manner and is left in the cask while the fuel is replaced.



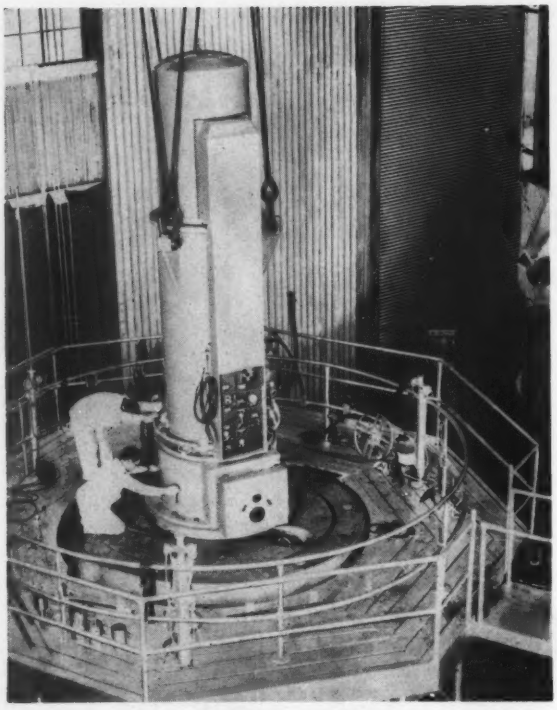
(Above) Cutaway view of rotary manipulator plug with periscope in position. (Right) Plan of plug.

Fuel Element Replacement

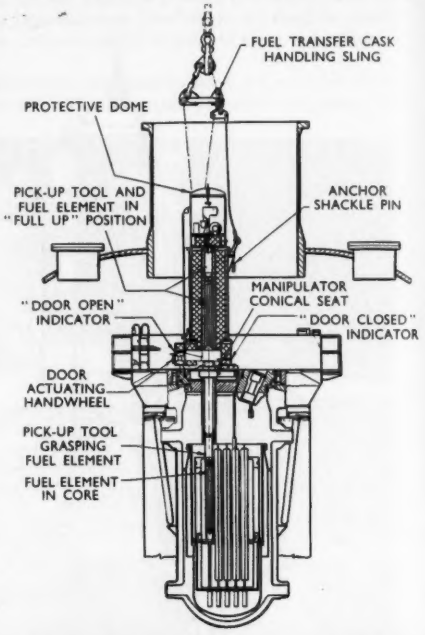
With the shield water still within the refuelling extension tank above the reactor vessel flange, the 67,000 pound rotary manipulator plug is lowered until it breaks the surface of the water



at which time the plug and water are lowered simultaneously until the plug is at rest on the reactor vessel flange protective ring. After the area wash-down is completed, the submarine lights and neutron detectors are inserted through holes in the plug, and connected to their service and instrumentation lines.

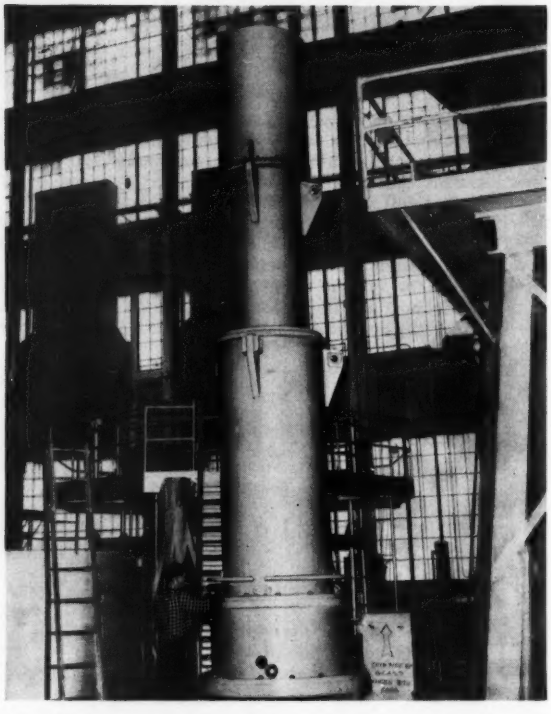


(Left) Positioning the fuel transfer cask on manipulator plug while slanted at 3.6°.


FUEL TRANSFER CASK HANDLING SLING
PROTECTIVE DOME
PICK-UP TOOL AND FUEL ELEMENT IN "FULL UP" POSITION
ANCHOR SHACKLE PIN
MANIPULATOR CONICAL SEAT
"DOOR OPEN" INDICATOR
"DOOR CLOSED" INDICATOR
DOOR ACTUATING HANDWHEEL
PICK-UP TOOL GRASPING FUEL ELEMENT
FUEL ELEMENT IN CORE

(Right) The fuel transfer cask in use.

(Below) the fuel transfer cask with the extension section for handling control rods.



The rotary plug refuelling port is then aligned over the first element to be removed by rotating the eccentric and concentric plugs to their predetermined positions and rotating the guide tube to its correct angular orientation with respect to that element. The guide tube extends from below the shield door in the eccentric plug to within a few inches of the control rod disconnect and handling knobs. Its purpose is to provide the missing link in a continuous guide system consisting of:

(a) The square tube that surrounds each element. (These "tubes" are interconnected by means of small H-beams that run the length of each tube between the tips of adjacent control rods that are located outside of the square tubes.)

(b) The square tube of the manipulator plug guide system.

(c) The square tube constructed within the refuelling cask.

With this continuous guide system the elements may be withdrawn and replaced without concern for the ship's motion and/or attitude. Guidance from the control rods is provided by notched fingers located within and at the bottom of the manipulator plug guide tube. The fingers, which will swing back to permit passage of a fuel element, have large throat lead-in surfaces to collect the control rod blade tips as they engage the notched blocks. On full scale mock-up tests, conducted at a 3.6° attitude, the blades entered the core in the exact centre of each control rod operating annulus.

After alignment is made over the fuel element to be removed (or, later, over the control rod to be replaced) a periscope is entered through a small opening in the plug door over the manipulator guide tube, and a visual check is made to ascertain that each setting has been accurately obtained.

The fuel transfer cask is then positioned on the manipulator plug and the plug and transfer cask doors are opened, permitting the grappling tool to descend. A visual check of the grasp can be made by means of the lead glass window located in the concentric plug of the manipulator. Two 1,000 watt underwater lamps provide the illumination. The lamps are enclosed within a Pyrex tube with screened bottom openings to permit the cooling water to pass by the lamp, yet retain glass particles in the event of lamp breakage.

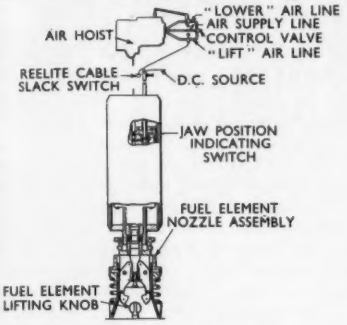
The chain-suspended grasping tool is designed to grasp the element on contact and automatically prevent further paying out of chain. If an improper grasp is made, the tool interlock will not permit hoisting of the element (or control rod). Once a proper grasp is made, a mechanical lock will prevent an accidental release of the load. As the element is hoisted into the cask, a primary spray system cools the element and fills the cask once the watertight shield door at the bottom is closed.

The manipulator shield door is closed, and the loaded fuel cask is transported to the service vessel, where it is unloaded in the reverse manner. The elements are stored under a multiport shield slab and in separate poisoned racks. While the cask is on the service vessel, the manipulator is repositioned for the next cycle.

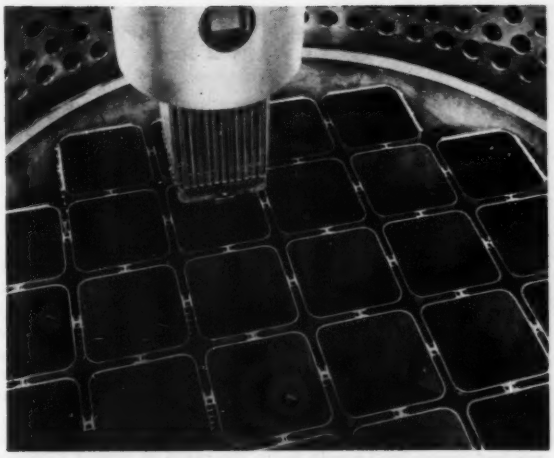
All elements are removed, followed by the control rods (if depleted). The fuel transfer cask is used for this task by adding a six foot long shield extension to the cask to provide the

necessary overall height for the long control rod-follower assembly.

New rods are then inserted, followed by two fresh elements. A special tool then places a 100 curie source into each of the two elements and the remaining fuel is loaded. Fresh fuel may be inserted with the transfer cask or with an adapter



(Left) The pneumatic grab for fuel element handling.



(Right) Dummy fuel element protruding from the manipulator guide tube.

utilizing manual guidance into the manipulator. After each element is loaded, the manipulator is relocated to a "home position" to allow readings to be taken with the two WL6376 neutron detectors.

Reactor Vessel Closure

After the manipulator is removed and stored, the upper grid plate and upper flow baffle are replaced by the internals transfer cask, followed by removal of the flange protective ring, and the installation of the reactor vessel head. Six studs are inserted and tightened and the seal weld membrane is hand welded. The remaining studs are installed and tightened with the tensioner to predetermined values.

The control rods are reconnected to the drive actuators, the refuelling tank seal removed, new instrumentation installed around the reactor vessel, and the head insulating cover installed. The system is now ready for start up tests, followed by achieving critically after the installation of various containment closure heads. The N.S. *Savannah* is then ready for voyages equal to over 1,200 days at average operating power.

APPENDIX

Containment Vessel Design and Construction

The containment vessel consists of a centre cylindrical section 35 ft diameter, 16 ft long, of thicknesses varying from 2⅝ in. to 3¾ in.; two hemispherical ends, 1.24 in. thick, attached to the cylindrical portion; and a cupola, 14 ft 6 in. outside diameter, 16 ft high on top of the cylinder section. The shell steel is SA-212, Grade B. Four internal stiffening rings are provided in each hemispherical section. The half-rings are made of a 1.25-in. thick web, and a 15×1.5 in. face plate. These connect to the top and bottom of two stiffening rings at the ends of the cylindrical section. The cylindrical section also contains two partial rings in the centre portion of the same size as hemisphere rings. Each end ring of the cylindrical section is made of high tensile steel with a web of 1¼ in. thickness and a 15 in. face plate which varies in thickness from 1½ in. to 3 in. The other rings are SA-212-Grade B. The vessel shell and rings weigh 275 tons and encompass a volume of 41,300 ft³.

The containment vessel is designed to withstand an internal pressure of 186 p.s.i. resulting from the maximum credible accident. In addition, it must support approximately 500 tons of radiation shielding (lead and polythene) on the outer surface of the shell under a 30° roll. There are no generally accepted codes or specifications written to cover a structure intended for such a dual purpose. For this reason, the shell system and shielding support structure were designed separately. The shell and all its penetrations comply in all respects with the A.S.M.E. Code for Unfired Pressure Vessels. The supporting rings have been analysed from first principles of mechanics. Members have been selected which provide a safety factor of four for all loads except that of the maximum credible accident, and three for all loads including the maximum credible accident. These safety factors are valid when shielding necessary for a 0.5 r/year radiation exposure has been applied to the containment vessel and at a 30° roll of the ship.

Shell thicknesses were determined from the A.S.M.E. Code for Unfired Pressure Vessels and checked in accordance with shell membrane theory. Membrane stresses for the design pressure were under 17,500 p.s.i. Principal stresses in the hemisphere shell due to external loading were checked and were not considered critical.

A detailed investigation of discontinuity points was made according to classical plate and shell elastic-energy techniques. Penetration reinforcement and reinforcement around 13 ft 6 in. dia. opening were designed to meet the requirements of the Marine Eng. Reg., USCG CG-115, dated March 1, 1956. An investigation was made where the hemispherical shell and cylindrical shell meet and where the stiffening rings and shell meet, for a maximum credible accident.

The governing condition for design of the containment vessel was the shell weight plus weight of shielding superimposed upon shell pressure stresses due to a maximum credible accident. Dynamic load factors of pitch and roll were included in deadweight determination.

Moments, shears, and axial forces on each ring were determined by application of strain-energy equations. The ring loading is reacted by the supports and transmitted to the ship structure. Hemisphere rings were analysed in the same manner with a saddle support at 45° to the vertical and restraint at top and invert of the end cylinder rings.

Preliminary investigation clearly indicated the critical design criteria for the containment vessel to be the dynamic 30° roll of ship in combination with a maximum credible accident and external shielding loads. The maximum stress was no greater than $F_t = 26{,}000$ p.s.i. and $F_c = 22{,}800$ p.s.i. for the combined loading analysis.

The containment vessel rests on three saddle-shaped longitudinal supports, port and starboard, welded to the tank top and one support under each hemisphere ring. Thirty-eight lubrite bearing plates ½ in. thick are provided to transmit loads to supports. The containment vessel is fastened to the ship structure only at the aft end of the cylindrical portion with twenty-four 2¼ in. diameter high-strength bolts. This connection is designed to resist rotation and forward and aft motion, but allows the vessel to expand freely due to temperature changes. Support at mid-height in way of collision mats prevents dislocation from rolling of the ship and the chocks at "B" Deck ensure that the containment vessel is not dislocated by a 90° heel of the ship.

Foundations for nuclear components are provided inside the containment vessel designed to transmit the loads through the shell to the external foundations in the ship structure.

The two hemispherical heads, the cylindrical section and the cupola were fabricated separately. All welds in each section were preheated and the roots inspected by the magnetic particle method. After completion of welding all butts and seams were 100% radiographed. All fillets were inspected by the magnetic particle method. Each section was then stress relieved and assembled in place on the ship. A jacking arrangement was devised for handling the 275 ton assembled vessel so that the bearing pads and tapered liners could be fitted to suit the shell.

The vessel is provided with one 13 ft opening at the top of the cupola, through which passes all the equipment to be fabricated and installed within the vessel. Two 42 in. access openings are provided on the vessel top, forward and aft of the cupola, with two standard manholes on the underside of the vessel. In addition, 76 other penetrations varying from 1 in. to 12 in. diameter are provided to accommodate piping and electrical wiring. The reinforcement for all penetrations was designed to comply with the A.S.M.E. Code for Unfired Pressure Vessels and Marine Eng. Reg. U.S.C.G. CG-115. Penetrations for piping smaller than 2 in. consist of a nozzle through which the pipe passes. For piping larger than 2 in., a nozzle with a reinforcing doubler is used. Electrical penetrations are also made through appropriate nozzles. The seal for wiring is epoxy resin. This seal has been tested successfully at the pressure and temperature of the maximum credible accident.

The containment vessel, after assembly, with no internals in place, was hydrostatically tested to a pressure equivalent to the maximum credible accident. The hydrostatic test consisted of filling the vessel with steam heated water (approximately 130°F) while venting all air pockets. When the vessel was full, the pressure was increased in 10 increments to 173 p.s.i. (a later calculation of the pressure produced in the maximum credible accident) by a feed water pump, keeping the minimum shell temperature at approximately 100°F to eliminate the possibility of brittle fracture.

In the event that the ship should sink, the two manholes in the bottom of the containment vessel have been arranged to open to prevent collapse of the containment vessel. The bolts securing the cover are necked down so as to break when the pressure outside corresponds to a water depth of 100 ft. When the pressures inside and out are equalized, a spring arrangement returns the cover to the closed position, thus preventing easy escape of the contained material.

The containment vessel is continuously air-conditioned to maintain a temperature of 120°F and is ventilated only when the hatch is open and the plant shut down. Continuous radiation monitoring of exhaust air is provided.

The SNAP Programme

U.S. AEC's Space-Electric Power Programme

By GUVEREN M. ANDERSON, Lt.-Col., U.S. Air Force
and FRANK H. FEATHERSTON, Lieut.-Cdr., U.S. Navy

PROJECT SNAP (Systems for Nuclear Auxiliary Power) is the U.S. AEC's programme for developing electric power generating systems for spacecraft using radioisotope and fission reactor heat sources. Power supplies which can operate for many months or years, as opposed to the several weeks expected from chemical systems, decrease the investment required for a given reconnaissance mission by reducing the number of vehicles which need to be put aloft.

Early feasibility studies indicated that for future payloads the required power level would be in the range of several kilowatts for some missions. Physical engineering limitations restrict the use of chemical, solar and radioisotope power systems to one or two kilowatts, leaving reactor systems as the only source of power for larger units.

The SNAP programme was formally inaugurated in the summer of 1955 when the U.S. AEC asked American industry to submit proposals for the development of nuclear auxiliary power units for satellite use. The AEC intended to explore the possibilities of using both radioisotope and reactor heat. The Nuclear Division of the Martin Company was awarded a contract to develop radioisotope power sources for space use; Atomics International won the competition to begin development of a compact reactor.

Radioisotope Development Programme

Beta-emitting waste fission products such as strontium-90, promethium-147 and cerium-144, and reactor-produced alpha-emitters such as curium-242, polonium-210 and plutonium-238 were considered suitable as fuels, but, principally because of its availability, cerium-144 was selected for the development of a radioisotope fuelled space auxiliary power unit. Conversion of the heat of decay of the cerium-144, which was to be contained in a right circular cylinder fuel block about 1 ft long and 4 in. in diameter, was to be accomplished by boiling liquid mercury and allowing the vapour to expand through a small turbine which turned an alternator. A power plant design specification of 500 We for a 60-day mission life was decided upon, and Thompson-Ramo Wooldridge was chosen under subcontract to the Martin Company to develop the mercury turbogenerator equipment. The programme was given the designation of SNAP–1. This effort was aimed at replacing battery systems, such as were used in Sputnik I, that had a life of only a few weeks.

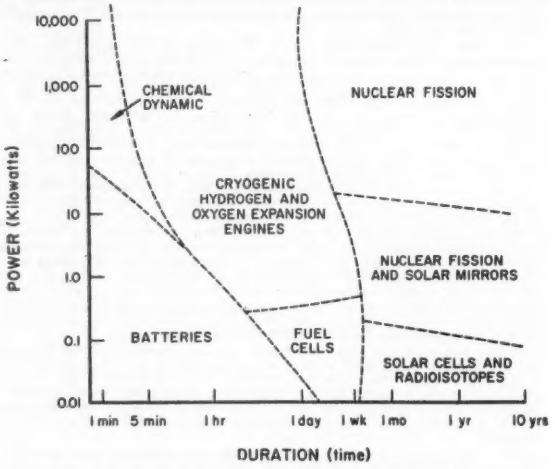


Fig. 1.—Applicability of space auxiliary power sources.

The choice of mechanical conversion equipment for SNAP–1 was dictated by an evaluation of the state of heat to electricity conversion technology at the time design specifications were chosen in 1957. Only the well-established technology of rotating equipment design and development appeared to be capable of producing working conversion equipment on the schedule that was effective. Even then, the design configuration chosen for the SNAP–1 conversion system implied that it was necessary to make a significant contribution to rotating equipment technology by specifying that all of the moving parts of the system be mounted on a common high speed shaft that would be lubricated by the mercury working fluid. SNAP–1 would employ no auxiliary systems for pumping working fluid or for lubricating bearing surfaces, thereby improving system reliability and compactness.

With a full development programme under way to produce a cerium-144 fuelled SNAP–1 unit utilizing rotating conversion equipment, several low level sub-contracts were, none the less, let through the basic Martin Company radioisotope contract to

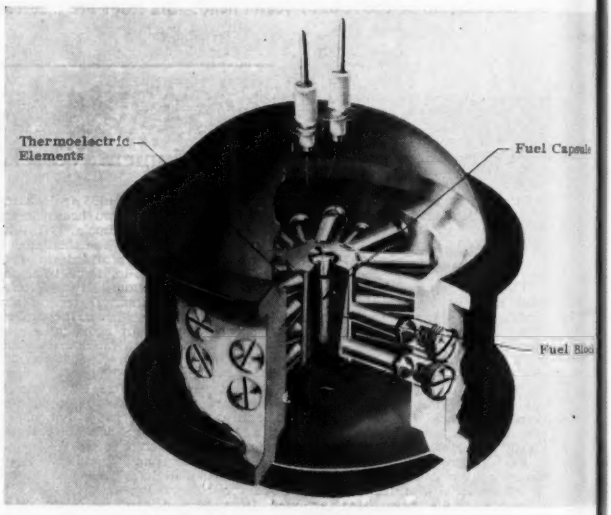


Fig. 2.—SNAP–3.

develop advanced thermoelectric and thermionic heat-to-electricity conversion devices that could be used with radioisotope heat sources. Generators of this type would be truly static and would not introduce perturbing torques to space vehicles that would require compensation. Thus, work was started in 1958 on two thermoelectric demonstration devices, one to be built by Westinghouse for delivery to Martin, the other to be built by Minnesota Mining and Manufacturing, also for delivery to Martin. Each company intended to use thermoelectric materials and fabrication techniques privy to itself. In addition to these thermoelectric devices, development of demonstration vacuum type thermionic units was initiated under Martin sub-contract by Thermo-Electron Engine and United Electronics. This whole programme to develop and demonstrate advanced energy conversion techniques was given the designation of SNAP–3.

The SNAP–3 programme was quick to yield results when Minnesota Mining and Manufacturing delivered a complete

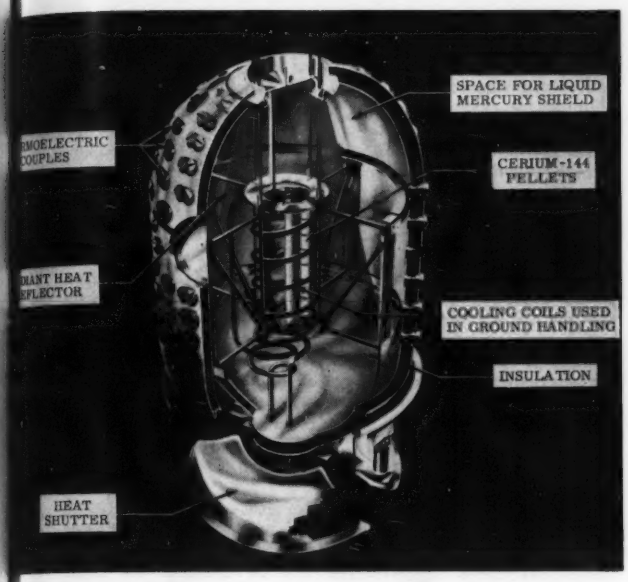


Fig. 3.—SNAP-1A.

thermoelectric generator to Martin in December 1958. A cut-away of this generator is shown in Fig. 2. The generator was designed to receive polonium-210 fuel capsules to be furnished by the AEC's Mound Laboratory. Martin assembled and tested the complete generator in early January 1959 and delivered the unit to the AEC. This proof-of-principle demonstration device produced some 2.5 W with a half charge of polonium-210 fuel and was first introduced to the world by President Eisenhower on January 16, 1959, as the SNAP–3 " atomic battery."

The successful demonstration of thermoelectric conversion in conjunction with a radioisotope heat source, as embodied in the SNAP–3 device, resulted in a re-orientation of the parent SNAP–1 programme in the spring of 1959. The re-oriented programme, now called SNAP–1A, uses cerium-144 heat sources similar to those developed for the SNAP–1 programme but dispenses with rotating conversion equipment. An artist's concept of this generator is shown in Fig. 3. The fuel sources are contained in the centrally located cylinder. A void space between the cylinder and the outer skin is used to contain liquid mercury for shielding purposes. Thermoelectric materials of the type used in the SNAP–3 device are used to accomplish the heat to electricity conversion. The individual couples are located in the outer skin. Such a unit in space can produce 250 W continuously for a year rather than 500 W for 60 days as was planned for SNAP–1. Development has been planned about a 125 W module, two of which would be used.

Development of the turboelectric machinery for SNAP–1 was concluded with the delivery by Thompson-Ramo Wooldridge of a completed turbogenerator package. The turboelectric technology developed under this programme was transferred to the reactor development portion of the SNAP programme where mercury conversion equipment of the same type was under development by the same company. The SNAP–1 turbogenerator package has since been tested with mercury vapour under design conditions and has operated far in excess of its test specifications. The AEC will complete the SNAP–1A development programme this year followed by a ground demonstration of a fuelled 125 W module in 1961. Radioisotope fuelled thermoelectric units of the SNAP–1A type will thus be available as essentially " off-the-shelf " technology for space or other uses. A number of prototype generators are also being developed for naval uses and for possible employment as power for a lunar probe.

Reactor Programme Development

In the 1955 competition sponsored by the AEC for the development of a space reactor system for satellite use, some thirteen proposals were received from which the reactor system proposals submitted by Lockheed Aircraft and Atomics International were chosen. Low level support of both of these reactor concepts was continued while the merits of the two received closer scrutiny. Finally the Atomics International concept was chosen in the spring of 1956 and was designated the SNAP–2 programme.

The SNAP–2 system employs a 50 kWt compact core reactor that uses liquid NaK as the reactor coolant to transfer heat to a mercury Rankine secondary loop to produce 3 kWe in a high speed shaft unit similar to that previously under development for SNAP–1. An electro-magnetic NaK pump has been added to the common shaft to circulate the primary loop fluid. The use of liquid metal coolant in combination with massively hydrided and homogeneously mixed zirconium-uranium fuel elements has permitted the development of a small volume (about 0.3 ft³) reactor core while still retaining essentially a thermal spectrum. A cut-away of the SNAP–2 reactor which weighs 200 lb is shown in Fig. 4. Reactor control is accomplished through the rotation of beryllium metal control drums located in the reflector surrounding the core. There is no piercing of the core by control rods. The complete space reactor-power conversion system will weigh approximately 600 lb and is being designed for programmed start-up in orbit. A thermo-mechanical control channel will be used for load control when the system is operating at designed power conditions.

Development of the SNAP–2 system has proceeded rapidly. A prototype core is undergoing tests at design power and temperature (1,200°F coolant outlet temperature) in a special test facility. As of August 11, 1960, this core had operated for 153,000 kWh and included a continuous full-power run of 1,000 hours duration.

Design and fabrication of this prototype test reactor was preceded by two years of research and development in studying the physics of the metal-hydride reflector controlled reactor system and in creating the technology of fabricating, hydriding and cladding SNAP–2 fuel elements.

Development of the power conversion equipment for SNAP–2 has proceeded in parallel with the reactor development. Design and component testing of successive developmental stages of turbines, alternators, pumps, shaft bearings, and their sub-assemblies has been accomplished. Several combined rotating units, the complete turbine-alternator package with the NaK and mercury circulating pumps on one shaft, have been designed, built and tested. Meanwhile an extensive programme is under way to determine the boiling and condensing heat transfer, corrosion and mass transport properties of the mercury working fluid. Based

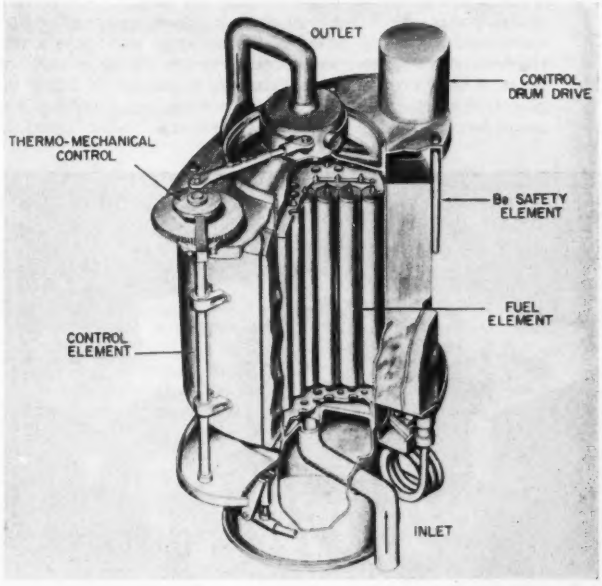


Fig. 4—SNAP–2 reactor.

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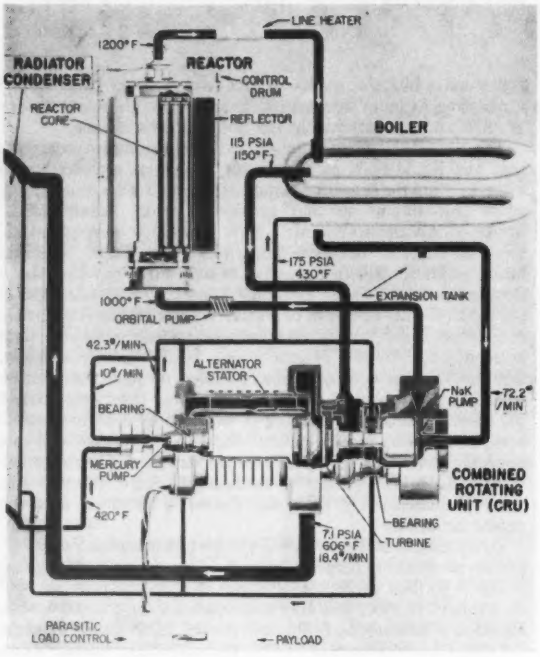


Fig. 5.—SNAP-2 APU system schematic.

on this work there is considerable testing and preliminary design work being done on the NaK/Hg boiler and the mercury condenser. Development of the condenser is particularly complex because this component must be structurally integrated into the space craft, configured to ensure positive fluid condensation under zero gravity conditions and be impervious to punctures by micrometeorites. A schematic of the SNAP-2 system which incorporates the turbo-alternator-pump package in a complete space power system is shown in Fig. 5.

There are several important areas that are peculiar to the development of complete nuclear-electric systems which need to be started automatically on orbit and with load following control equipment which is optimized for reliability and low weight. A nuclear-electric power system that must start up automatically and operate unattended for a long period of time in space must have a very reliable control system, and adequate system testing is vital. A series of three evolutionary system tests is planned to follow the prototype reactor core test that is currently under way. These are to be conducted in an especially constructed facility called the SNAP Environmental Test Facility (SETF) which is now complete; general details are shown in Fig. 6. It will be possible to conduct full nuclear tests in one of two power test vaults while the radioactive components from a previous test in the other vault are being removed for examination, the vault decontaminated and, then, made ready for another test.

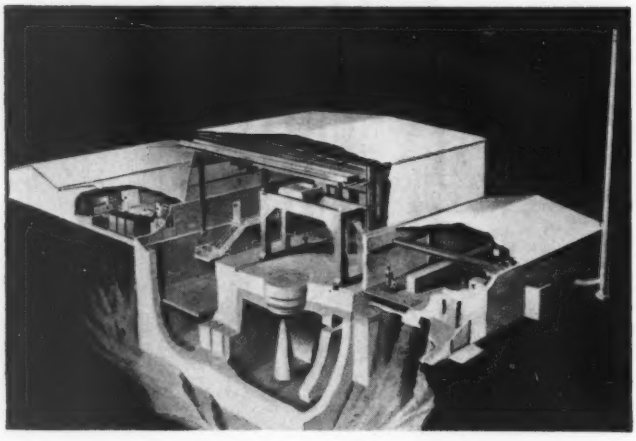


Fig. 6.—SNAP Environmental Test Facility.

The first of the tests in the facility, S2–DS–1 (SNAP–2 Developmental System–1) will employ a SNAP–2 reactor separated from system components by adequate shut-down shielding. The objective is to gather system control and component response data under design cycle conditions. Control and response data of this type can be acquired only during full system operation and is necessary before work can proceed on final refinement of component designs. S2–DS–1 will begin in November 1960 and will continue for approximately a year.

The next complete system test, S2–DS–2, will begin in the summer of 1962 and will test a SNAP–2 system in its final space vehicle configuration with the reactor closely coupled to the system components. The first phase of S2–DS–2 calls for an extensive period of non-nuclear testing in order to gain the maximum information while still having direct access to the equipment. Once this is complete, full nuclear tests will begin, to confirm the performance of the system.

The third of the evolutionary system tests is S2–PS–1 (SNAP–2 Package System–1) scheduled to start in the summer of 1963. S2–PS–1 will be a ground demonstration of a flight prototype system (S2–FS–1). A prototype will be available for flight demonstration.

Because the SNAP–2 system must give unprecedented performance in terms of being capable of long term reliability in an environment that is completely new to power plant development, the three-step programme is regarded as the minimum necessary. It is estimated that in spite of the complexities the three SNAP–2 system tests will cost under $2 million each: a very reassuring figure when one considers the expenses usually associated with regular stationary reactor experimentation and testing.

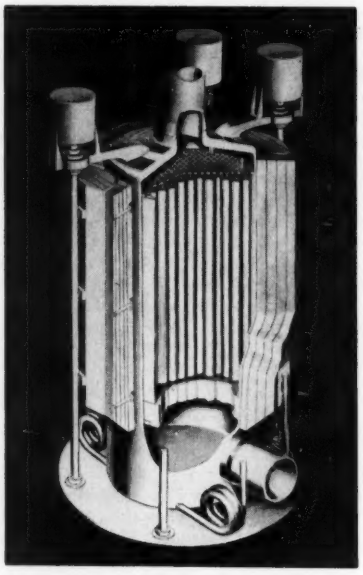


Fig. 7.—SNAP–8 reactor.

Further Developments

Until January 1959, the 3-kWe SNAP–2 unit was the only space nuclear reactor auxiliary power system under development by the AEC. Since that time two additional space power unit development programmes have been started, based on the metal hydride reactor technology of SNAP–2. These are SNAP–8, a 30 kWe unit and SNAP–10, a 300 kWe unit.

SNAP–8, designed to weigh 300 lb and shown in Fig. 7, will be developed in conjunction with the National Aeronautics and

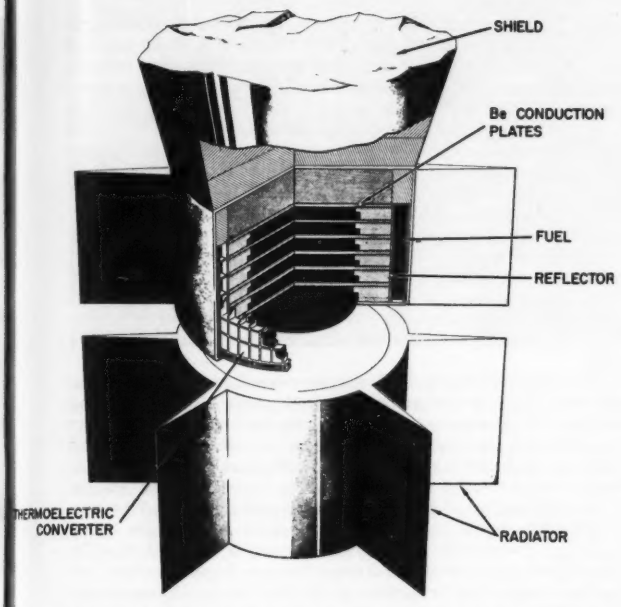


Fig. 8.—(Left) SNAP-10 reactor the core of which first went critical on August 2 at the AEC's Santa Susana facility operated by Al.

Space Administration (the NASA). The AEC will extend the SNAP-2 reactor concept to 600 kWt (from its present design level of 50 kWt) to provide the thermal capability to run one or two 30 kWe combined rotating units at one time. The NASA has entered into a separate contract with Aerojet General to develop the power conversion equipment and to accomplish over-all system integration. In many respects SNAP-8 is a scaled-up version of SNAP-2.

Study in 1959 revealed the feasibility of constructing a conduction cooled thermoelectric conversion space reactor auxiliary power unit using SNAP metal hydride fuel and the lead telluride solid state conversion materials employed in SNAP-3. Such a unit would produce about 300 W in a 350 lb package with no moving parts. A cut-away is shown in Fig. 8. The reactor's strong negative temperature coefficient alone is sufficient to provide control for a one- to three-year lifetime. Control drums are, therefore, not employed in this reactor.

This power plant, known as SNAP-10, will fit the power and weight requirements of several satellite missions planned in the next few years. Work is currently under way to develop solar cell arrays in this power range but the technical difficulties that are being encountered indicate that SNAP-10 may be the only way of providing reliable power that can be independent of vehicle attitude and sun-to-shade storage problems. The SNAP-10 programme is pointing towards early flight demonstration of a 300 W APU.

Table 1. SNAP System Specific Weights.

System	Electrical Power, kW	System Weight, lb	Shield Weight for Present Transistors, lb (approx.)	Specific Weight, lb/kW e
SNAP-10	0.3	355	250 (at 6 ft)	2,000
SNAP-2	3.0	600	250 (at 13 ft)	280
SNAP-8	30	1,200 (est.)	300 (at 20 ft)	50
SNAP-8	60	2,500 (est.)	500 (at 20 ft)	50

The zirconium-uranium hydride fuel element material that makes possible the small thermal cores of the SNAP reactor systems constitutes a significant step forward in reactor technology. The compactness of a comparable fast reactor system would be achieved at a much greater fuel inventory than that required for the SNAP reactor. Thus, SNAP-2 and -10 will use slightly under 3 kg of enriched uranium, SNAP-8 about 5 kg.

Growth of the family of SNAP reactor systems aptly illustrates that specific power is inversely proportional to power level. The weight of the reactor and shielding dominates in the 300 W SNAP-10 system while a much greater portion of the overall system weight of SNAP-8 is attributable to the radiator. The weight of shielding for transistorized payloads contributes no more than 10 lb/kW at the power level of SNAP-8 (Table 1). From the foregoing it is apparent that the typical temperature limits of the metal hydride system yield a lower limit for specific power, in the region of 50 lb/kW.

Studies are now under way for a 300 kWe Rankine cycle system which may achieve a specific weight of 15–20 lb/kW. Beyond this, advanced thermionic reactor systems give promise of yielding specific weights less than 10 lb/kW. Such systems are predicted to be available in the next decade. Recent emphasis on the demonstration of electric propulsion for interplanetary travel by the NASA points to the need for lighter systems if one hopes to achieve reasonable transport times.

Advanced Systems—
A Personal Appraisal

By DR. ALVIN M. WEINBERG, (*Director, Oak Ridge National Laboratory*)

THERE is no agreement among reactor experts as to what are "advanced" civilian power reactor concepts. A reactor system which to most of the world appears extremely difficult is, to its proponents, beset only with usual developmental problems. Nevertheless, one can define, somewhat arbitrarily, "advanced" civilian power reactors as those which significantly exceed current technology either in respect to temperature, simplicity of fuel recycle, or neutron economy (breeding). Each of these objectives is predominant in what, to my mind, are the three most important trends in advanced reactor development:—

(1) The trend toward unclad or ceramic-clad—and therefore "dirty"—solid fuel element systems.

(2) The renewed interest in high-temperature liquid fuels.

(3) The continued work on fast and slow neutron breeders.

It is not surprising that most of the questions concerning these reactors centre around the fuel element or the fuel liquid.

Ceramic-clad or Unclad Solid Fuel

The incentive to replace the metallic cladding on the fuel element comes from the desire to raise temperature, improve neutron economy, and simplify the fuel recycle. The price which must be paid is the necessity for handling radioactivity either in the main circulating system or at least in a purge system.

Four gas-cooled reactors now under active design provide for handling fission products in the coolant helium. These reactors are the Philadelphia Electric HTGR (40 MWe), the OEEC Dragon (20 MW thermal), the Krupp Pebble Bed Reactor (15 MWe) and the Los Alamos Turret (3 MW thermal). All of these reactors are graphite moderated and are "semi-homogeneous"—i.e., the fissile uranium and fertile thorium are intimately mixed with the graphite. The essential advantage of this arrangement is that the fissile material, being greatly diluted by graphite, can be operated at very high heat ratings,

viz., 500-1,000 kW/kg of fissile material. In point of fact the HTGR, Dragon and PBR all have specific powers of 1,000 kW/kg, the Turrent only 430 kW/kg. The absence of metallic cladding allows exit gas temperatures of the order of 1,400°F to 2,400°F (this last being the helium exit temperature of Turret) and overall thermal efficiencies of 35% or better. The high temperature and high heat ratings also result in compactness; ultimately, such reactors should be fairly cheap to build.

The most obvious problem—fission products in the gaseous stream—can be met in two ways: by living with a dirty system, as in the Turret reactor, or by cladding the fuel elements with nearly impermeable graphite and coping with the residuum of fission products which leak through the coating by means of a fission gas trapping system.

The development of "impermeable" graphite, especially in U.K., has been astonishing. It has been reported, for example, that graphites are now available which will pass only 10^{-6} cm^3/sec of gas through a cm^3 of graphite under a pressure difference of one atmosphere. The designers of the HTGR expect to be able to keep the total activity of the coolant gas below 100 c, which will be a remarkable achievement as the equilibrium gaseous and volatile fission product activity in the fuel elements of HTGR will be about 100×10^6 c. All reactor engineers will be most interested to see how well the fission product trap and the impermeable graphite cladding work in the HTGR when it starts operation, scheduled for 1963.

The other central uncertainty of the semi-homogeneous graphite reactor is the limitation of fuel lifetime imposed by reactivity losses. To calculate the lifetime of a fuel loading requires taking small differences of large numbers, and this is always an uncertain business. The importance of this uncertainty may be judged by the following example. Large reactors of this type are supposed to achieve one to two fissions per initial fissile atom without recycling. At one fission per initial fissile atom, the burn-up cost is about 2.5 mill/kWh, which is quite low. However, with no recovery of fissionable material, this cost is inversely proportional to the burn-up; if only 1/2 fission per initial fissile atom is achieved, the burn-up cost becomes 5 mill/kWh, which is higher than the fuel cycle cost of the pressurized-water systems. There is enough uncertainty in the capture cross sections of the fission products, and in the values of η in the epithermal region of U^{233}, to make a fully reliable estimate of the achievable burn-up very difficult and, unfortunately, the economics of such reactors are very sensitive to the burn-up.

Heat transfer tests in an autoclave on fuel elements for General Atomic's HTGR at San Diego.

If the burn-up should prove to be disappointingly low, the economic situation could be retrieved if the reconstitution of the fuel were very cheap. This does not seem to be particularly easy to achieve. To burn the graphite, as has been suggested, is not attractive as the very large volumes of radioactive CO_2 which result from combustion of graphite are awkward and expensive to handle. Experiments at ORNL have shown that the impregnated UO_2 can be leached from pulverized graphite with hot, concentrated HNO_3. The recoveries in experiments to date have been about 99%. It is, however, too early to give reliable estimates of the fuel recycle costs for such graphite fuels.

An attractive variant of the semi-homogeneous solid fuel element graphite system is the BeO moderated, gas cooled reactor. This system has better neutron economy than the graphite moderated reactor but, so far as I am aware, too little has been done on the properties of BeO to evaluate the reactor fully. Active work on such systems is being done by the Australian Atomic Energy Commission and by General Atomic.

All of these gas cooled semi-homogeneous reactors use highly enriched U^{235} which, as the reactor runs, is replaced by U^{233}. The very long-term outlook depends upon improving the conversion ratios and rationalizing the fuel cycle. Undoubtedly very advanced graphite or Be moderated gas cooled systems can be made with conversion ratios of better than 0.9. However, to combine high conversion ratio with very high specific power will require extensive manipulation of fuel under power; that is, the fuel becomes semi-labile, and to work out reliable on-stream shuffling mechanisms will require much development. It is for this reason that the Turret experiment, which addresses itself to a rational handling of fuel under power, will be watched with great interest by those concerned with gas-cooled breeding reactors.

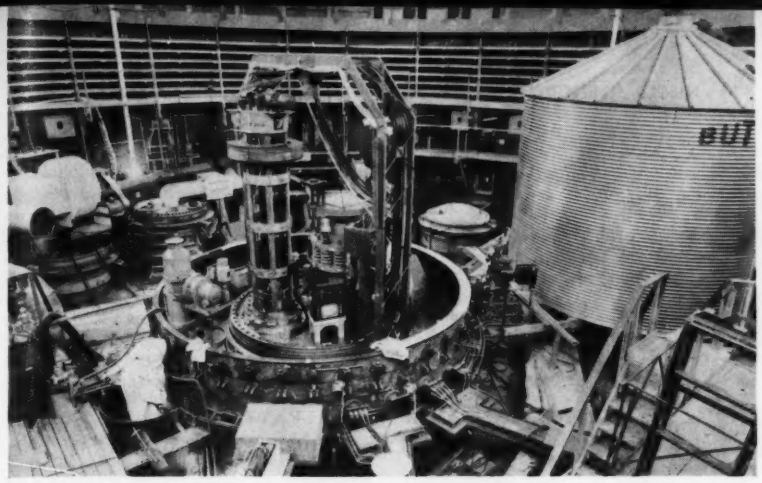
High-temperature Liquids

From the dirty, semi-labile solid fuel element reactor to the circulating fuel fluid reactor is a short step. The more the fuel is reshuffled in a solid fuel reactor, the more the problems of the reactor resemble those of the liquid reactors—difficult maintenance, stringent requirements of leak-tightness, complicated inventory control.

Of the three liquid fuel reactors—aqueous homogeneous, uranium-bismuth, and molten fluoride—the last seems to have the fewest remaining developmental problems. This may sound surprising as, by and large, the molten fluoride technology, having been developed for application to nuclear-powered aircraft, is not widely known. Yet the total amount of money which has been devoted to molten fluorides as reactor fuels—including a molten fluoride reactor, the Aircraft Reactor Experiment—is of the order of $75 million over the past 10 years. That the technology is not widely known does not mean that it has not been widely explored.

There are now available a series of fluoride mixtures—for example, LiF-BeF-UF_4-ThF_4 and a nickel-molybdenum alloy, INOR-8—which are compatible up to temperatures of at least 1,400°F and under radiation. The salts are also compatible with graphite. Such remaining questions as the requirement of keeping oxygen out of the salt (to avoid precipitation of UO_2) or the problem of keeping the salt above its melting point (~800°F) or the feasibility of reactor maintenance can be answered only by a reactor experiment. A small reactor experiment—5,000 kW thermal—is now under design at ORNL, and it is hoped that it will be finished by 1963. The reactor will be moderated with graphite, the salt being in direct contact with the graphite. The whole circulating system will be made of INOR-8, and the system will be constructed so as to simplify maintenance.

The great advantage of the molten salt system is that it allows one to burn any of the fissile materials—U^{233}, U^{235}, Pu^{239}—at very high thermal efficiency, and with extremely high burn-up. As thorium is soluble in the melt, it should be possible to achieve rather high conversion ratios (hopefully above unity) with a technology that is the same for core and



Interior of Enrico Fermi fast reactor containment building showing reactor, sodium pumps and heat exchangers.

blanket. The combination of very high thermal efficiency, high conversion ratio, and very high burn-up gives to the molten salt system the distinction of having perhaps the lowest calculated fuel cycle cost of any reactor—of the order of 0.5 mill/kWh, including inventory charges. Whether the maintenance costs and the capital charges for this system will more than offset the extremely low fuel cycle cost remains to be seen. Experience with the forthcoming molten salt reactor experiment will certainly clarify this crucial point.

The unclad gas cooled and the molten salt reactors are intended primarily to achieve very low fuel cycle costs in systems which burn uranium or plutonium. By incremental improvements, both of these systems might eventually achieve breeding, the molten salt having a significant edge over the gas cooled solid fuel systems in this respect.

The other approach to breeding is to try to develop, *ab initio*, a technology which is specifically aimed at breeding, rather than to develop breeders from reactors which start as non-breeders. The two reactors of this class are the sodium cooled fast Pu and the aqueous homogeneous thorium breeders.

Fast Breeders

Most of the world's work on breeders is devoted to the Na cooled fast breeder. There are now, either in operation or under construction, five fast breeders—EBR-I, EBR-II, Enrico Fermi, Dounreay, and Obninsk (USSR). The first four reactors are fuelled with U^{235}, the last one with PuO_2. Although some delays have been experienced in getting Dounreay and Fermi on the line, there is no essential doubt that fast reactors can operate at high total power. The real question, however, is the amount of burn-up achievable with the fuel elements now available. The first core fuel for the Fermi reactor—a uranium-molybdenum alloy—is expected to withstand less than 1% burn-up. As the ratio of U^{235} to U^{238} in the fuel is 1 : 3, this means that each gram of fuel must be recycled 25 times before it is completely burned. If each recycle—including chemical and metallurgical handling—costs as little as $1/gram, the fuel recycle cost alone comes to $25/gram which, at 30% thermal efficiency, amounts to about 4 mill/kWh. It is this essential cost which must be greatly reduced before the fast breeder can be considered a serious economic possibility.

Several schemes for increasing the burn-up or reducing the chemical costs have been proposed. Perhaps most significant are the proposals to use oxide fuels; it is significant that the Obninsk reactor is fuelled with PuO_2. Burn-ups of 13% are considered feasible with oxide fuels, but these high burn-ups are achieved only at the expense of a lower breeding ratio.

Another scheme for improving the burn-up without going to oxide is the use of "fissium"—i.e., an alloy of Pu containing fission product elements in about the proportion they appear in fission. Such alloys are supposed to tolerate radiation better than does the unalloyed fuel, but even here there are problems from the build-up of fission gas pressure. It is suggested that the resulting swelling be eliminated by venting the fuel pins to the cooling stream. This would, in some of the suggested embodiments, introduce fission products to the coolant and one is then confronted with many of the problems of the circulating fuel reactors.

The use of non-aqueous processing—slagging and zone melting, for example—is suggested as a means of greatly simplifying the fuel recycle and thus reducing the economic penalty of frequent fuel reprocessing. The EBR-II has an integrated, non-aqueous fuel reprocessing plant connected with it. Its operation will be tremendously important to a future assessment of fast breeders.

The fast breeder is generally considered the ultimate hope of nuclear energy. Yet my own opinion is that, up till now, the central problem of the fast breeder—the fuel recycle—has received relatively little attention, let alone been solved. Until much clearer evidence is at hand that the fuel can be recycled economically, I think it is a serious error to depend exclusively on fast breeders as the ultimate energy source.

Aqueous Homogeneous

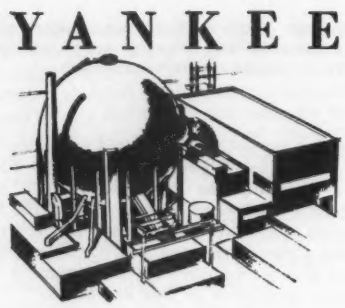
The other breeder is the aqueous homogeneous. The very extensive high-powered runs of HRE-2 (the longest run lasted 105 days) have verified the belief that an aqueous homogeneous reactor can be operated for long periods before critical items of equipment fail. Moreover, the maintenance of HRE-2 has proved surprisingly practical. On the basis of the experience with HRE-2, it is estimated that the maintenance cost of a very large (300 MWe) homogeneous reactor would exceed that of a similar-sized heterogeneous pressurized-water reactor by no more than 0.5 mill/kWh. As the overall fuel cycle cost of the aqueous homogeneous breeder could be as low as 1 mill/kWh, it is argued that the aqueous homogeneous breeder could be the basis for an economic, asymptotic energy system.

The main problem which remains to be solved is the stability of the fuel. It has been found in HRE-2 that at high power densities and under conditions of poor flow, U^{235} precipitates onto the zirconium core tank. This converts the core tank into a fuel element of undetermined composition and, in the course of operation of HRE-2, two holes were burned in the core tank, presumably where uranium had precipitated. The holes are now being patched (remotely, as the radiation level at the holes is still 100,000 r/h) and the direction of flow is being reversed in an attempt to keep the wall swept clear of debris. With these changes it is hoped that the fuel will be stable at the designed core power density of 17 kW/l.

The other major problem of HRE-2—the thorium blanket—seems to the proponents of the system to be capable of solution either by use of ThO_2 slurry or ThO_2 pellets. The reason for this optimism is that operation of HRE-2 to date has shown that a dirty, high-pressure liquid system is feasible. Hence there is hardly any question that a *stationary* bed of bare ThO_2 pebbles, bathed in D_2O, will be satisfactory and to this extent the blanket is available. However, there is some question as to how much motion of the pebbles is tolerable as attrition might lead to build-up of fines which could cause plugging in the rest of the system.

It is, of course, difficult for me to be objective with respect to the aqueous homogeneous reactor. Nevertheless, I am much impressed by the fact that HRE-2 operated for long continuous periods, and it is my belief that the problem of fuel stability—which looms now as a major difficulty of aqueous homogeneous reactors—will prove to be only an incidental difficulty.

In closing these few remarks, I must reiterate what has been said often, but which remains to be said again. The full promise of nuclear energy will be fulfilled only if we learn how to burn the essentially infinite reserves of low-grade uranium and/or thorium. This means that development of breeder reactors, rather than being an unimportant, long-haired exercise, is really the central problem of nuclear energy. I believe that the whole nuclear development will take on a more rational aspect if and when we realize that this is its central business and, in consequence, devote to this central business a much greater share of our resources than we now do.



YANKEE Atomic Electric

by JAMES N. KILPATRICK
(Atomic Power Dept., Westinghouse Electric Corporation

General Description

WITH a net electrical capacity of 134,000 kW, the Yankee Atomic Electric plant at Rowe, Massachusetts, will be the first large-scale nuclear plant to generate electric power for civilian use in the New England area. The plant is a joint venture of 10 utilities and will transmit its power output into each of their systems over the existing New England transmission system grid. The sponsoring utilities who, at present, represent over 80% of the electric output of the New England states, organized the Yankee Atomic Electric Company in 1954.

Westinghouse Electric Corporation was selected to design and furnish the nuclear steam generator, and Stone and Webster Engineering Corporation was responsible for designing the balance of the plant and its total construction. The U.S. AEC participated in this project to the extent of a $5 million contract to Yankee for the necessary research and development and by waiving interest charges on the nuclear fuel for a period of five years. The total capital requirements of the Yankee project have been estimated at $57 million which includes cost of the first core, land, financing costs, civil works, initial start-up expenses and working capital.

Rowe was selected as the plant site for several reasons. Situated in the northwest corner of Massachusetts and only three miles south of the Vermont border, Rowe is in a sparsely settled area. The nearest privately owned and occupied dwelling is nearly a mile away and within a five-mile radius the population density is only 25 persons per square mile. The Yankee plant has been erected on the eastern shore of Sherman Pond, which also serves the Sherman hydro-electric station of the New England Power Company. Construction at the site began on April 19, 1958, the 150-ton pressure vessel was raised into position in February, 1960, and the core was loaded in July, 1960. Initial criticality was achieved on August 19, 1960.

YANKEE ATOMIC ELECTRIC

Boston Edison.	Hartford Electric Light.
Central Maine Power.	New England Electric System.
Central Vermont Public Service Corp.	New England Gas and Electric Ass.
Connecticut Light and Power.	Public Service Co. of New Hampshire.
Eastern Utilities Associates.	Western Massachusetts Electric.

Outline Design

The heart of the Yankee plant is a closed-cycle light water cooled and moderated reactor, using enriched UO_2 pelletized fuel encased in stainless steel. An average temperature of 514°F is maintained in the coolant, which is constantly circulated under 2,000 p.s.i.g. pressure. The primary coolant passes through four parallel loops to vertical steam generators and produces steam on the secondary side at approximately 467°F and a pressure of 500 p.s.i.

The entire main coolant system, as well as the high temperature-high pressure portions of the primary plant

(Below) Universal handling tool developed by Westinghouse for Yankee.

(Right) Installing core barrel assembly in reactor.

auxiliary systems, are located inside a 125-ft spherical steel envelope, this vapour container being designed to contain the pressure build-up resulting from any accidental major loss of coolant.

The expected full load turbine heat rate at 1.5 in. Hg back pressure is 11,345 B.t.u./kWh at 145 MW load. The average auxiliary power requirement at full load is 9,335 kW, and the expected full load station heat rate is 12,270 B.t.u./kWh. Although the secondary plant is designed primarily for the 145 MW rating, design margins permit a gross turbine generator capability of 160 MW. Provision is made for complete steam dumping in the event of a generator cut-out at full power.

Control

The plant is operated from a central room in the turbine building. The reactor itself is controlled, either manually or automatically, by means of Ag-In-Cd rods (with followers) and although this mechanical system is capable of achieving complete subcriticality of the reactor core, a supplementary chemical control system based upon boron injection is provided to ensure cold shut-down should this prove necessary. In any case a small quantity of highly concentrated solution is to be injected into the main coolant system during cold plant shut-down procedures. In the unlikely event of a rupture accident, resulting in a loss of moderator-coolant, a low concentration boric acid solution would be charged into the main coolant system and would provide core cooling and negative reactivity, thereby minimizing the consequences of the accident.

During normal shut-down decay heat is removed by an auxiliary cooling system. The nuclear instrumentation monitors the neutron flux in the neutron shield tank surrounding the reactor vessel. This system indicates start-up rate and flux level for all power ranges, and scrams or shuts down the core when the start-up rate or flux level is too high. Similarly, the reactor is shut down when a low flow or low pressure condition exists in the main coolant system. Pump failures, with ensuing decrease or loss of coolant flow, can be controlled without dangerous power or reactivity excursions. Should failure occur in a single pump, stability can be regained with practically no increase in temperature level, even without scram. In failures of two or more pumps, however, a low flow scram occurs and keeps the reactor under control.

Primary Coolant Control

To preclude bulk boiling, a saturation over-pressure is always maintained. Plant load changes affect the temperature of the primary coolant and, therefore, its expansion and contraction. These density changes during normal plant load changes are accommodated by the expansion and contraction of the steam volume in a pressurizer. Coolant volume changes which cannot be accommodated in this way result in the blowing of safety relief valves on the pressurizer vessel during increasing volume surges, or the scramming of the reactor during decreasing volume changes. The main coolant system is essentially a constant high pressure system.

A high degree of purity is maintained to reduce the rate of activity build-up in the plant. Irradiated corrosion products are the principal contributors to long-lived activity, but the corrosion control holds general corrosion to a minimum by limiting the dissolved oxygen concentration. In addition, a purification system removes undesirable impurities from the coolant.

During plant operation, and even while shut down, a component cooling system services the main coolant pumps, low pressure surge tank cooler, and neutron shield tank. This system also serves the shut-down cooler during shut-down, and the fuel pit cooler, sample cooler, and waste disposal system as required.

A system of valves and headers in the vent and drain system facilitates draining of the primary plant for maintenance purposes. The main coolant system (except for the reactor vessel) is drained to the primary drain collecting tank, where it may be dumped to the waste disposal system.

Effluent Treatment and Disposal

The waste disposal system receives, contains, adequately treats and safely disposes of all radioactive wastes in such a manner as to yield concentrated liquid wastes and ashes. Low activity level liquid wastes are discharged to the river after dilution with secondary plant effluent cooling water. Gaseous waste, after suitable dilution with fresh air, is dispersed to the atmosphere in small quantities and only under favourable meteorological conditions.

Some airborne activity may be present in the vapour container atmosphere after depressurization of the main coolant system. To permit entry into the container under these conditions, a purge system of filter banks and heating coils dilutes the air in the container.

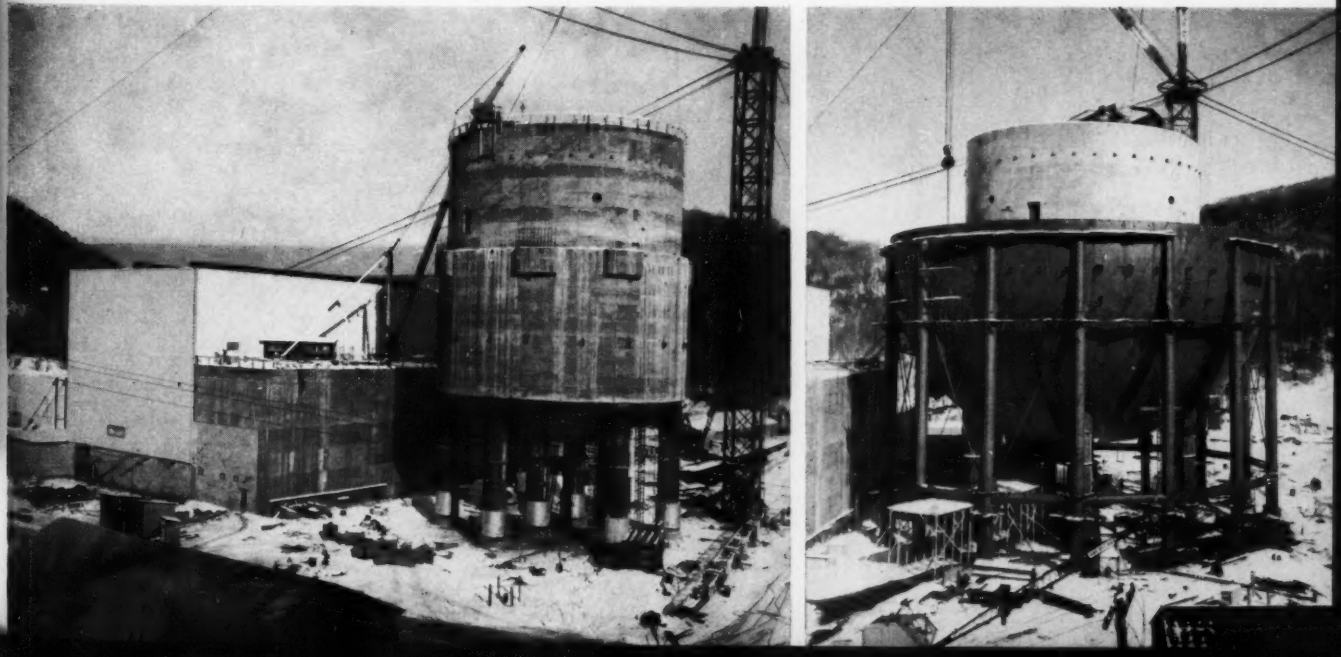
The Yankee plant will be shut down periodically for refuelling, maintenance, and inspection. Because those systems which contain the moderator-coolant may become increasingly radioactive during plant lifetime, this contamination must be reduced where immediate access is desired. The decontamination system provides for the decontamination of isolated main coolant system loops, as well as of the pressurizer vessel.

In order to refuel the reactor without hazard to personnel, facilities are provided for remote underwater removal and transfer of spent fuel assemblies from the reactor vessel to the spent fuel storage pit. Installation of new fuel assemblies is also performed remotely and under water.

Area Monitoring

An operational radiation monitoring system is provided to detect, compute, and indicate the radiation level at selected locations inside and outside the vapour container. If these levels should exceed predetermined values, alarms would be

Fig. 1.—Stages in the construction of Yankee (Dec., 1958; Mar., 1959).



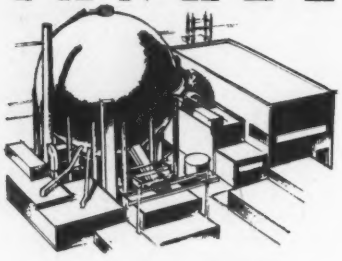
sounded. This radiation monitoring serves a dual purpose: it warns of any radiation health hazard which might occur, and it gives early warning of any plant malfunction which, if unchecked, might result in a health hazard or in plant damage. The Yankee plant vapour container air particle detector, for example, continuously samples and checks the air for the presence of particulate matter. Air is drawn in through a section of moving filter paper upon which the particulate matter is collected. Directly behind the filter paper, at the point where collection occurs, is a scintillation counter. The complete air particle detector consists of two units: the air particle detector itself, and a vacuum pump. These units, mounted in a weatherproof enclosure on the outside surface of the vapour container, can operate between −25 and +145°F and withstand an internal pressure of 45 p.s.i.

This air particle detector has a sensitivity of 10^{-9} to 10^{-6} $\mu c/cm^3$ to beta radioactive particulates detectable in less than 15 min with a fixed filter or in 30 min with the filter tape moving at the rate of 1 in./h. A fast advance of speed of 6 in./min makes it possible to clear contaminated tape from the detecting area within a very short time. The detector is calibrated as necessary by a remotely controlled radium foil source. Visible and audible alarms operate when: (a) The air flow rate becomes too high or too low; (b) the filter tape breaks, or fails to advance in the normal manner; (c) the valve closure circuit is energized.

The site and portable radiation detection equipment provides the necessary information to detect any increase in normal background radiation at the plant site. Personnel dosimetry and film badges are also provided to measure the integrated exposure of all personnel with access to, and working in, contaminated areas. As further protection, a radiation detector which monitors both hands and feet simultaneously is located at the entrance to the plant lunch-room.

Two site monitors provide for continuous measurement of air particulates, and are in addition to those mounted in the main plant stack discharge. One of these site monitors is installed approximately one-half mile below the plant site, the other being located at the Harriman hydro-electric station, approximately four miles above the Yankee plant area. Both monitors employ the continuously moving filter paper principle, with a continuous air flow of 10 ft^3/min through the filter to ensure adequate sensitivity. The health physics department at Yankee is responsible for ensuring that all prescribed safety measures are observed, and the conducting of radiological measurements and surveys.

YANKEE



Reactor Vessel

by P. BLAIR HAGA

(Supervisor, Systems Engineering, Westinghouse Atomic Power Department

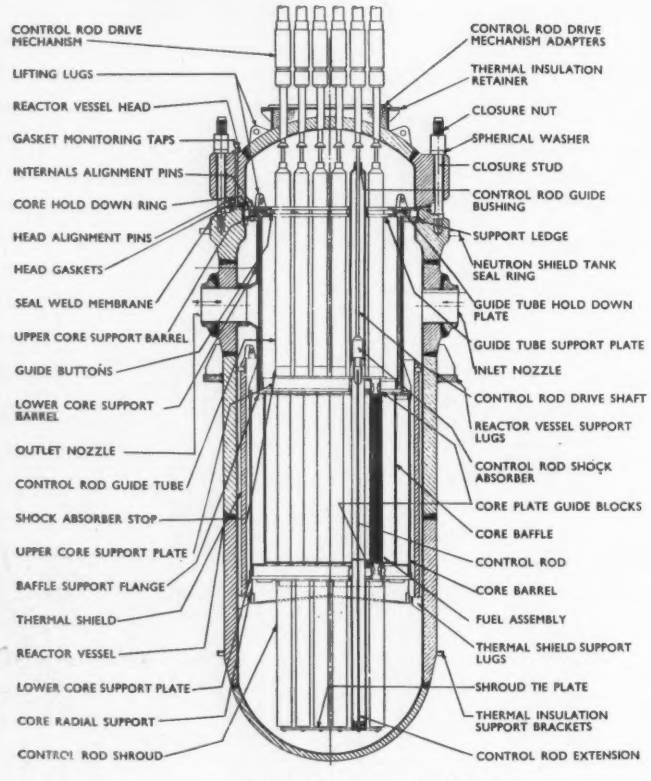


Fig. 2.—Reactor nomenclature.

AS can be seen in Fig. 2, the reactor vessel is cylindrical with a permanently attached hemispherical bottom head and a removable, spherically dished, top head. Primary coolant water enters the vessel through four inlet nozzles located above the core, passes down through the reflector and thermal shield regions, up through the core, and leaves the vessel through four outlet nozzles located in the same horizontal plane as the inlet nozzles.

All primary coolant nozzles are located above the core to provide a container which can be readily filled with coolant to cover the core, if a rupture occurs in the primary system. A penetration of the reactor vessel itself was not considered to be feasible. Although this arrangement of nozzles gives a higher system pressure drop than with inlet nozzles at the bottom, the added safety features were considered to be more important. A section through the nozzles is shown in Fig. 3. It can be seen in Fig. 3 that the inlet plenum is sealed from the outlet plenum by a tight fit between the reactor vessel and the lower core support barrel. This seal is effective at operating temperatures, because of the differential expansion between the stainless-steel barrel and the carbon-steel vessel.

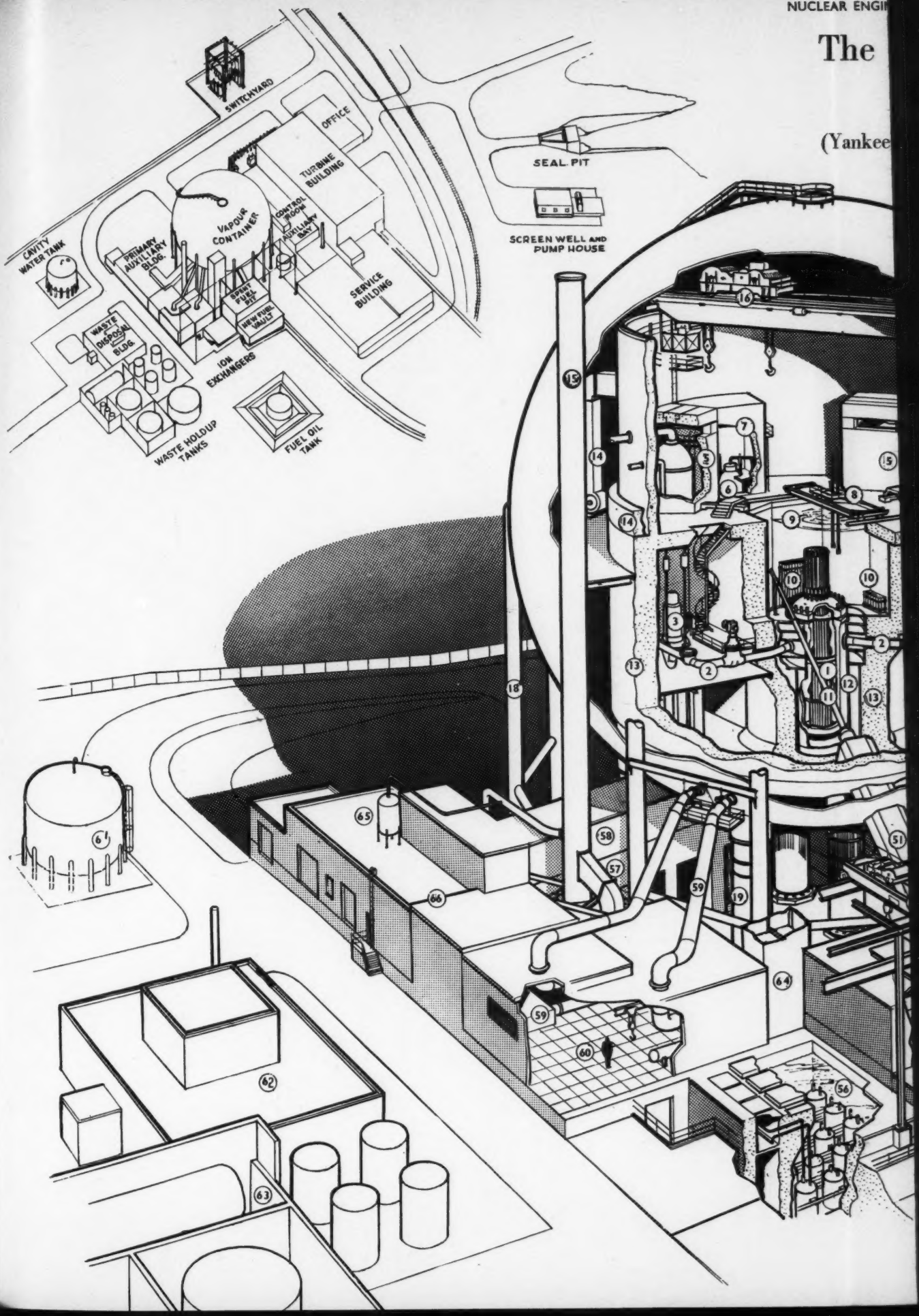
A small flow of primary coolant passes from the inlet plenum up through holes in the flange on the lower core support barrel and holes in the hold-down ring, down inside the guide tubes and mixes with the coolant, leaving the core. This flow maintains the head and vessel flanges at the same temperature and reduces the cooling requirements for the 24 control-rod-drive mechanisms mounted on the closure head.

With the exception of the thermal shield, all internals are supported by a ledge cut in the vessel flange and keyed to the vessel and head by four pins. All internals are held in place by the compression loading of the core hold-down ring by the closure head.

The thermal shield is designed to reduce thermal stress in the vessel wall and fast neutron irradiation. Fig. 4 shows curves of temperature and thermal stress distributions in the vessel wall opposite the centre of the core. The thermal stress is conservatively low as the shield thickness was chosen

The

(Yankee


SWITCHYARD
OFFICE
TURBINE BUILDING
SEAL PIT
SCREEN WELL AND PUMP HOUSE
CONTROL ROOM
AUXILIARY BAY
VAPOUR CONTAINER
CAVITY WATER TANK
PRIMARY AUXILIARY BLDG.
SPENT FUEL PIT
NEW FUEL VAULT
SERVICE BUILDING
WASTE DISPOSAL BLDG.
ION EXCHANGERS
WASTE HOLDUP TANKS
FUEL OIL TANK

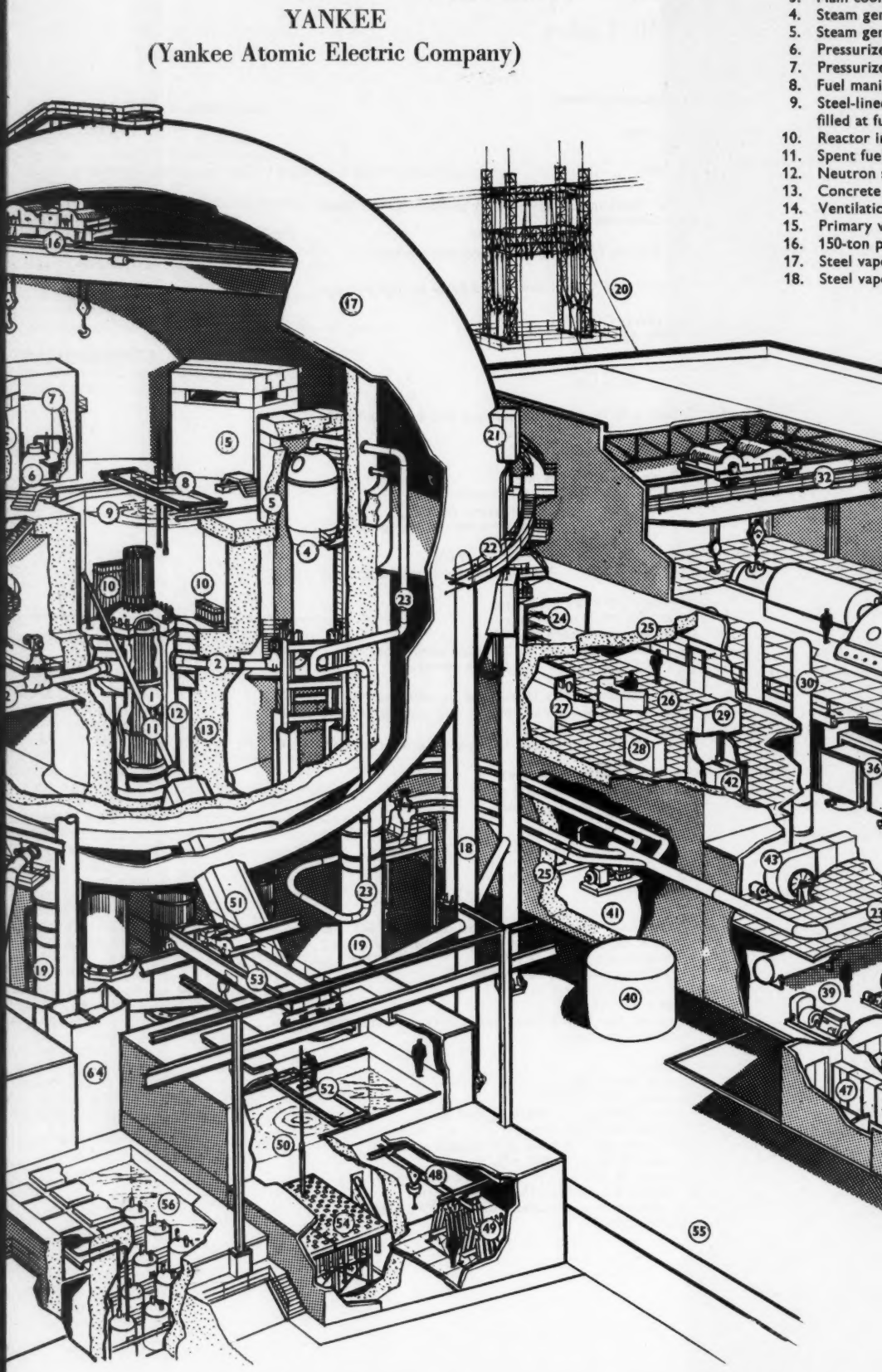
The World's Reactors

No. 30

YANKEE

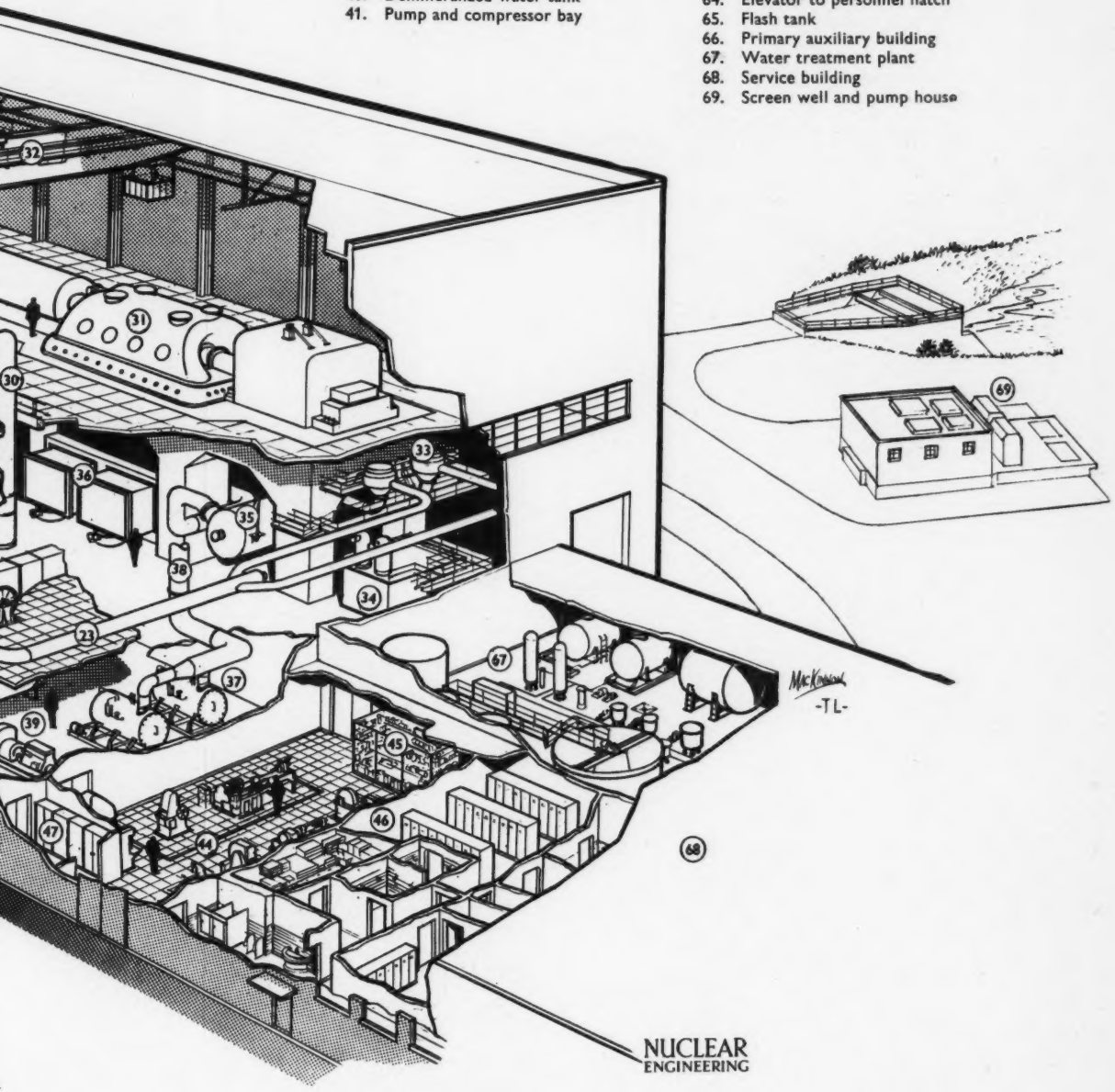
(Yankee Atomic Electric Company)

1. Reactor c
2. Main cool
3. Main cool
4. Steam ge
5. Steam ge
6. Pressuriz
7. Pressuriz
8. Fuel mani
9. Steel-line
filled at fu
10. Reactor i
11. Spent fue
12. Neutron
13. Concrete
14. Ventilatio
15. Primary v
16. 150-ton p
17. Steel vap
18. Steel vap



KEY

1. Reactor core
2. Main coolant loop
3. Main coolant pump
4. Steam generator
5. Steam generator compartment
6. Pressurizer
7. Pressurizer compartment
8. Fuel manipulator crane
9. Steel-lined shield tank cavity—water filled at fuel change
10. Reactor internals storage
11. Spent fuel chute
12. Neutron shield tank
13. Concrete biological shield
14. Ventilation and heating ducts
15. Primary vent stack
16. 150-ton polar crane
17. Steel vapour container
18. Steel vapour container columns
19. Steel encased concrete columns
20. Switchyard structure
21. Electrical penetrations
22. Walkways
23. Main steam piping
24. Cable tray room
25. Concrete shielding
26. Control room
27. Main control board
28. Valve and safety injection panels
29. Radiation monitor panel
30. Feed water heaters
31. Turbine generator
32. Turbine room crane
33. Throttle valves
34. Turbine oil reservoir
35. Interstage moisture separator
36. Main condensers
37. Heating boilers
38. Heating boiler stack
39. Engine driven electric generator set
40. Demineralized water tank
41. Pump and compressor bay
42. Control room elevator
43. Ventilation fan and ducts
44. Machine shop
45. Tool crib
46. Toilet and locker facilities
47. Equipment cleaning
48. New fuel storage vault
50. Spent fuel pit
51. Concrete shield for spent fuel chute
52. Spent fuel manipulator crane
53. Work area crane
54. Spent fuel storage
55. Spur tracks
56. Ion exchangers
57. Shielded radioactive pipe tunnel
58. Non-radioactive pipe tunnel
59. Purge ducts
60. Ventilation room
61. Safety injection water storage tank
62. Waste treatment building
63. Waste hold-up tanks
64. Elevator to personnel hatch
65. Flash tank
66. Primary auxiliary building
67. Water treatment plant
68. Service building
69. Screen well and pump house



NUCLEAR ENGINEERING

The World's Reactors

No. 30 Yankee

TYPE: Thermal heterogeneous (PWR).

LOCATION: Rowe, Massachusetts.

PURPOSE: Power production.

OWNER: Yankee Atomic Electric Company (combination of 10 New England utility companies).

DESIGNERS: Westinghouse Electric Company and Stone and Webster.

STATUS: Critical: August 19, 1960. Full power scheduled for late autumn.

OUTPUT:
First core: 392 MWt; 110 MWe.
Future: 482 MWt; 134 MWe.

FUEL:
Enriched uranium oxide (3.4%).
Total loading: 23,738 kg UO_2.

CORE:
Cylindrical prism with 76 channels, arranged 4–6–8–10–10–10–10–8–6–4.
Equivalent diameter: 75.68 in. (1.92 m).
Height: 91.86 in. (2.33 m).
Average power density: 58.4 kW/litre.

hot:
- ε = 1.06.
- p = 0.562.
- k_{eff} = 1.090.
- τ = 50.2 cm².

k_{eff}, cold, clean, 1.219.

Excess reactivity for:
Temperature: 9.7%.
Xe and Sm (at rated power): 3.2%.
Burn-up: 9.0%.
Total excess reactivity 21.9% $\Delta k/k$.

Neutron flux:
Thermal, average: 2.0×10^{13} n/cm², sec.
Thermal, max.: 9.6×10^{13} n/cm², sec.
Temperature coefficients at zero power: (design calculation, zero power.)
Moderator: -3×10^{-2} % $\Delta k/k°F$.
Fuel: -4.6×10^{-3} % $\Delta k/k°F$.

FUEL ELEMENTS:
Rod bundle type, arranged 18×18 to form an element 7.66 in. (194.5 mm) square, with shortened rows at two corners to leave space for cruciform control rods.
No. of rods per element: 304 or 305, UO_2 pellets clad in stainless steel.
Overall dia. of rod: 0.340 in. (8.64 mm).
Thickness of cladding: 0.021 in. (0.533 mm).
Diameter of pellet: 0.294 in. (7.47 mm).

CONVERSION FACTOR: 0.529.

BURN-UP: Average: 7,800 MWd/te.

CONTROL:
Silver-indium-cadmium rods of cruciform section, 7.865 in. (199.8 mm) square, and 0.265 in. (7.24 mm) thick, with Zircaloy-2 followers. Active length: 90 in. (2.286 m).
No. of rods: 32 (24 regulating; 8 fixed shim).
Total worth of rods: 15% $\Delta k/k$.
Scram time: less than 2 sec.
The reactor has no burnable poison, but a boric acid injection system is included in the design.

COOLANT:
Pressurized water.
Inlet temperature: 499°F (260°C).
Outlet temperature: 532°F (278°C).
Coolant pressure: 2,000 p.s.i.a. (140.6 kg/cm² abs.).
Mass flow: 37.8×10^6 lb/h (17,140 te/h).
Channel velocity (average): 14 ft/sec (4.27 m/sec).
Heat transfer area: 15,500 ft² (1,440 m²).
Heat flux: Average: 86,300 B.t.u./ft², h (6.50 cal/cm², sec).
Maximum: 446,000 B.t.u./ft², h (33.6 cal/cm², sec).
Maximum fuel element temperatures: Fuel: 4,330°F (2,388°C).
Cladding: 663°F (350°C).

PRESSURE VESSEL:	Carbon steel, clad with stainless steel. Height: 31 ft 6 in. (9.6 m). Inside diameter: 9 ft 1 in. (2.77 m). Thickness of side: 7.875 in. (200 mm). Design pressure: 2,500 p.s.i.a. (175.77 kg/cm²) at 650°F (343.3°C).
HEAT EXCHANGE:	Four primary loops each with one heat exchanger of the vertical shell and inverted U tube type. Heating surface: 4×13,430 ft² (1,247 m²). Steam generation: 4×500,000 lb/h (228 te/h). Steam pressure: 500 p.s.i.a. (35.15 kg/cm² abs.).
SHIELDING:	Radial shielding (total): 15 in. (0.381 m) of steel; 4 ft (1.524 m) water; 10 ft (3.048 m) concrete. Top: 7 in. (0.178 m) steel; 12 ft 9 in. (3.886 m) water.
CONTAINMENT:	Steel sphere. Diameter 125 ft (38.1 m). Thickness 0.875 in. (22.2 mm). Design pressure: 34.5 p.s.i. (2.425 kg/cm²).

Pull-outs already published in this series in "Nuclear Engineering" are:

No. 1. BEPO (April, 1956)
No. 2. CP5 (May, 1956)
No. 3. NRX (June, 1956)
No. 4. DIMPLE (August, 1956)
No. 5. ZEUS (September, 1956)
No. 6. CALDER HALL (October and December, 1956)
No. 7. RUSSIAN 5 MW (November, 1956)
No. 8. DIDO (January, 1957)
No. 9. THE SOUTH OF SCOTLAND ELECTRICITY BOARD STATION (February, 1957)
No. 10. BERKELEY POWER STATION (March, 1957)
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No. 12. DOUNREAY FAST REACTOR (June, 1957)
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No. 14. RWE 1 (September, 1957)
No. 15. LIDO (November, 1957)
No. 16. PLUTO (April, 1958)
No. 17. MERLIN (May, 1958)
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No. 23. ZENITH (November, 1959)
No. 24. G2 (and G3) (December, 1959)
No. 25. OMRE (February, 1960)
No. 26. DRAGON (July, 1960)
No. 27. BR-3 (August, 1960)
No. 28. N.S. SAVANNAH (October, 1960)
No. 29. DRESDEN (October, 1960)

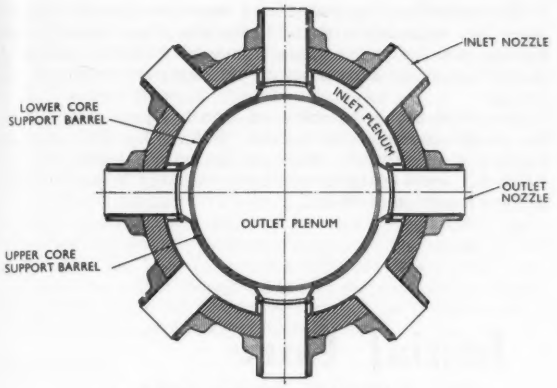


Fig. 3.—Section through nozzles.

to reduce fast neutron irradiation to what is considered to be an acceptable level. There are, however, uncertainties in the calculation of the fast neutron flux and samples of the vessel material have been inserted in the reflector region so that during regular refuelling operations these samples can be removed for testing to determine if any shift in transition temperature has occurred.

The joint between the vessel and closure head is designed to utilize either gaskets or seal welds to effect leak tightness, see Fig. 5; connections are provided to monitor any leakage past either gasket. The gaskets are self-energizing, stainless-steel O-rings plated with silver to provide good seating. The vessel will be operated initially with gaskets in order to obtain running experience. If leakage proves to be low, no seal welding will be employed.

The closure head is attached to the vessel with 52 5¼-in. studs. These studs contain a central axial hole suitable for heaters and extend through the closure nut for applying stud tensioners as an alternative. A reduced-diameter closure joint was selected to cut down the weight and cost of the vessel and closure head. This choice improved vessel flange design and reduced the required bolting force, but it followed that the thermal shield must be fabricated in segments. These segments are bolted together inside the vessel.

The weight of the reactor vessel is transmitted through 28 support lugs to a ring girder which is an integral part of the neutron shield tank. The ring girder rests on the concrete primary shield surrounding the vessel and neutron shield tank. Each support lug rests on a fitted, radial pin for accurate location of the vertical centreline of the vessel. The pins permit radial expansion of the vessel while maintaining fixed centreline and levelness.

In order to obtain insurance on the plant, it was necessary to design, fabricate, and test the reactor vessel in accordance with the American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code*, Section VIII. Although the reactor vessel is designed for a maximum system pressure of 2,500 p.s.i.a., it operates at a nominal system pressure of 2,000

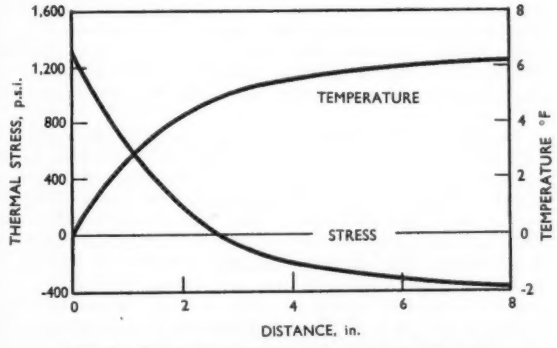


Fig. 4.—Temperature and stress in the vessel walls.

Table 1. MATERIALS

Heads and shell courses	SA-302 Grade B
Sheet cladding (0.109 in.)	SA-240 Grade S, type 304 stainless steel
Weld deposit cladding	Type 308L stainless steel
Flange forgings	SA-105 Grade II
Primary coolant nozzles	SA-182 modified to requirements of SA-302 Grade B
Control rod drive mechanism nozzles	Co-extruding tubing, base metal—SA-106 Grade C, cladding—SA-213, type 304 stainless steel
Thermal shield and internal barrels and plates	SA-240 Grade S, type 304 stainless steel
Closure studs	SA-193 Grade B-16
Closure nuts	SA-194 Grade 2HB
Closure washers	Special nitriding alloy G-135

p.s.i.a. The margin between nominal operating pressure and design pressure allows for system pressure transients and safety and relief valve settings. The vessel and head are exposed to inlet coolant temperature, which should be no higher than 513.5°F. (average temperature at no load), but additional margin for temperature transients is provided, as the design temperature is 650°F.

Only materials permitted by the ASME Code were used to fabricate the vessel. The internal surfaces of the vessel are clad with austenitic stainless steel; the base material is carbon steel. A partial listing of materials is presented in Table I. For the materials selected, the cylindrical section of the vessel is 7⅞ in. thick with 0.109 in. cladding; the hemispherical bottom

Table 2. CALCULATED WEIGHTS

Reactor vessel	300,000 lb
Closure head (without CRD mechanisms)	79,100 lb
Thermal shield	39,000 lb
Studs, nuts and washers	23,100 lb

head is 3¾ in. thick with 0.109 in. cladding, and the closure head is 7 in. thick with 0.109 in. cladding. The internal height of the vessel is 31 ft 6 in., with an inside diameter of 9 ft 1 in. The inside diameter of the primary coolant nozzles is 19½ in. Weights are given in Table II.

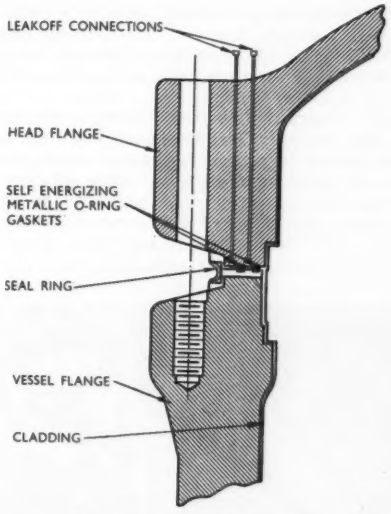


Fig. 5.—Gaskets and seal ring.

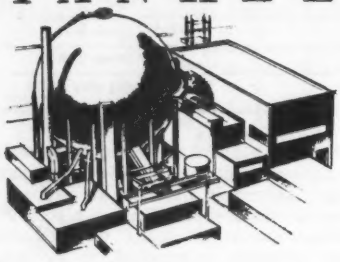
The only operational limits imposed on the vessel, other than maximum operating pressure and temperature, concern heating and cooling rates and minimum hydrostatic test temperature.

If the vessel is heated from an ambient temperature of 70°F, the maximum rate of heating must not exceed 50°F/h up to a temperature of 250°F. From a temperature of 250°F up to operating temperature the maximum rate of heating must not exceed 150°F/h. The maximum permissible step increase in coolant temperature is 40°F, when the vessel is in an isothermal condition.

The vessel may be cooled at a maximum rate of 150°F/h down to a temperature of 150°F; however, there may be some leakage past the gaskets when a rate of 50°F/h is exceeded. Actual operation is necessary to determine the extent of the leakage.

The vessel cannot be subjected to a hydrostatic test unless the metal temperature is at least 90°F. This limit was set by adding an adequate safety margin to the temperature at which the vessel material possessed a Charpy V-notch impact energy absorption of 30 ft lb.

YANKEE



Initial Core

by ARTHUR G. THORP, II

(Manager, Reactor Engineering Section, Reactor Development, Westinghouse Atomic Power Department

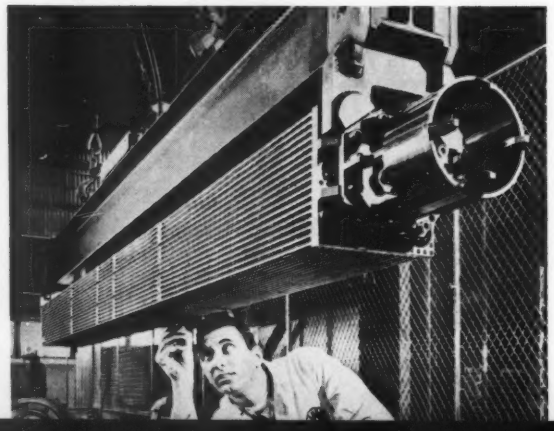
YANKEE'S initial reactor core, as designed by Westinghouse, can be approximated to a right circular cylinder 75.4 in. in diameter and 91.86 in. high, giving a length-to-diameter ratio of 1.2. The core consists of 76 vertical fuel assemblies, 24 control rods and eight fixed shim rods. The fuel assemblies are essentially square in cross-section and are placed in a close-packed square lattice. Fig. 7 shows a cross-section of the core.

The 76 individual, replaceable fuel assemblies are held in the core between the lower and the upper core support plates. Holes are provided in both support plates for the handling sockets which position the fuel assemblies and act as coolant inlets and discharge nozzles. These support plates are also provided with 32 cross-shaped slots to allow passage of the 24 cruciform control rods and the eight cruciform fixed shim rods. The axis of the control rods is parallel to the vertical axis of the core and each control rod is actuated by a separate mechanism above the core. For refuelling purposes the control rod drive mechanisms are disengaged from the control rod drive shafts, leaving the rods completely inserted in the core. After removal of the reactor vessel head, on which the drive mechanisms are mounted, and the reactor internals above the core, the fuel assemblies and control rods can be removed individually.

The reactor core is surrounded by a form-fitting baffle which confines the coolant flow within the fuel bearing zone. Outside the baffle the core barrel extends downward from the upper support plate, carrying the lower core support plate and the weight of the core. Coolant water between the form-fitting baffle and the support barrel acts as a neutron reflector to the core.

The core construction is of the "open lattice" type which permits flow redistribution throughout the core and assures that, even under the most unfavourable conditions which can be reasonably postulated, coolant reaches all portions of the core. The core is designed to operate with subcooled boiling during normal operation and the attendant density variations cause small radial components of velocity.

Fig. 6.—Fuel element.



Fuel Assemblies

The basic fuel assembly for the first Yankee core measuring 7.61 in. square and 111.25 in. overall (Fig. 6), is composed of 0.340 in. o.d. stainless steel tubes which contain uranium dioxide in the form of ceramic pellets 0.294 in. in diameter and 0.6 in. long. The required dimensional tolerances on diameter for the individual pellet are obtained by centreless grinding after sintering. Enrichment is 3.4 wt. % U^{235}. The fuel tubes are Type 348 stainless steel with a wall thickness of 0.021 in. Collapse tests have indicated that the strength of Type 348 is superior to other stainless steels in this application and that the can will withstand anticipated operating conditions without collapse.

The fuel pellets are placed in the tubes in groups of 25. Each group is separated from the next by a perforated stainless steel disc coated with braze material. Each disc has a circumferential groove into which the tube is crimped during the loading process, fixing its position until the entire sub-assembly is brazed. The discs are so spaced that each group of 25 pellets is allowed a free expansion space. The discs are provided in order to prevent progressive distortion of the fuel which might be caused by repeated differential expansion between an uninterrupted fuel column 90 in. long and the surrounding stainless steel tube; such differential expansion would occur with every change in reactor power level. Stainless steel end plugs are welded into the ends of each tube. The welding is of the tungsten inert gas shielded arc type[1]. Production welding has shown a high degree of integrity.

The fuel rods are assembled on a square lattice with a normal centre-to-centre pitch of 0.422 in. and on an odd pitch of 0.454 in. for those rows of tubes in line with the control rod vanes. Two fuel assembly designs are used which contain 304 and 305 fuel rods respectively; each is basically an 18 by 18 rod square. Rods are omitted from this pattern as required to provide slots for passage of the vanes of the cruciform control rods, thereby reducing the number of rods from 324, the number in the full 18 by 18 complement, to 304 and 305. The total number of rods which form the 76 fuel assemblies is 23,142.

The fuel rods are assembled into fuel element sub-assemblies by brazing[2] tubular spacers at 8 in. intervals to provide proper spacing between rods and to furnish the required structural rigidity. Thus the structural support for the fuel assembly is in the fuel tubes themselves. The fuel assembly consists of eight sub-assemblies which are free to float axially about a ninth centre sub-assembly attached securely by means of captive machine screws to the end plates. The eight outer sub-assemblies are guided at each end by end plugs which engage a hole in a slip-fit arrangement in the end plates. All sub-assemblies are tied together at a number of points along their length by means of tie straps, forming a complete fuel

assembly that functions as an integral unit. Rubbing straps brazed to the outer tube row of each of the outer eight sub-assemblies at four axial positions prevent undesirable rubbing of a control rod on a row of fuel tubes, and eliminate fuel rod surface contact between adjacent assemblies.

The Yankee fuel assembly is specifically designed to prevent excessive thermal bowing. Under the worst conditions of bowing caused by thermal gradients, allowing for the worst combination of mechanical tolerances, it has been calculated that there is adequate clearance between fuel assembly and adjacent control rod. Theoretical results were based upon linear and non-linear distribution across the fuel assembly and agree to within 5% with various full-scale experimental tests conducted on a fuel sub-assembly. A more extensive account of the fabrication of the assemblies will be published in a subsequent issue of *Nuclear Engineering*.

Control Rods

The reactor control rods are cruciform in shape, 24 in number, and are fabricated from a silver-indium-cadmium alloy (80% Ag, 15% In, 5% Cd). This alloy has a relatively high neutron absorption cross-section, is essentially "black" to thermal neutrons, and has a wealth of resonance structure to enhance its worth. The Ag-In-Cd alloy has approximately the same worth as hafnium and is available at a lower cost.

TABLE 3. MECHANICAL AND THERMAL DESIGN DATA

Total Heat Output, B.t.u./h	1.338×10^9
Total heat output, MW	392
Coolant flow	
Total rate, lb/h	37.8×10^6
Heat transfer rate, lb/h	34.0×10^6
Flow area in fuel rod cross section, ft²	15.4
Velocity along fuel rods, ft/sec	14.0
Pressure, p.s.i.g.	
Operating pressure, normal	2,000
Operating pressure, maximum	2,300
Operating pressure, minimum	1,800
Design pressure	2,500
Total drop across vessel	34
Drop across core	16
Heat transfer	
"Active" surface, ft²	15,500
Average flux, B.t.u./ft² h	86,300
Maximum flux, B.t.u./ft² h	446,000
Average film coefficient, B.t.u./ft² h °F	6,050
Burnout flux : $\left[(Q/A)_{BO}/(Q/A)_{max}\right]_{POB}$*	2.3
Bettis correlation, B.t.u./ft² h	9.80×10^5
Thermal conductivity of UO_2, B.t.u./ft h °F	1
Temperature, °F	
Coolant in main coolant system, Av.	514
Coolant in Fuel bearing portion of the core, Av.	516
Coolant at inlet to vessel	499
Coolant rise in core, Av.	33
Coolant rise in vessel, Av.	30
Film drop, Av.	14.3
Clad surface, max.	663
Centre of fuel, max.	4,330
Outlet of hot channel	603
T_m at heat exchanger	37
Steam temperature	475
Steam pressure, p.s.i.g.	525
Design hot channel factors	
F_Q, heat flux	5.17
F_θ, film drop	7.36
F_T, coolant rise	3.36
General	
Total core cross sectional area, ft²	31.0
Equivalent core diameter, ft	6.3
Length-to-diameter ratio of core	1.2
Length-to-diameter ratio of a flow channel	260
Fuel rod	
Outside diameter, in.	0.340
Tube wall thickness, in.	0.021
Total number of rods	23,142
Fuel length per rod, ft	7.5
Rod lattice, in.	0.422 and 0.454
Rods per fuel assembly	304 or 305
Total number of fuel assemblies	76
Fuel pellet	
Diameter, in.	0.294
Height, in.	0.60
Fuel assembly	
"Active" length, in.	91.86
Overall length, in.	111.25
Sectional dimensions, in.	7.61 × 7.61
Weight, lb	890
Movable Control rods	24
Fixed shim rods	8

Dimensions are for the initial core "cold" unless otherwise stated.
*POB = point of burnout.

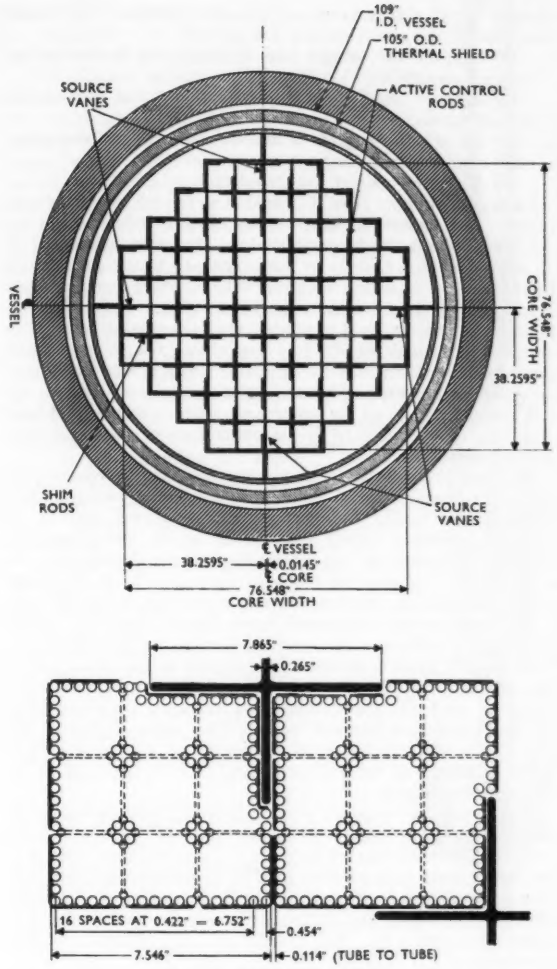


Fig. 7.—Core cross section.

As the corrosion resistance of Ag-In-Cd is marginal, even under the normal low oxygen conditions in the Yankee reactor, the control rods were plated with nickel to a nominal thickness of 0.0005 in. Heat treatment followed plating to give a diffusion bond between the Ni and Ag-In-Cd and to increase the creep strength.

A Zircaloy-2 follower, connected to the active portion of the control rod, acts as a guide and prevents the formation of a "water slot" in the core when the control rod is withdrawn. The follower also prevents excessive by-pass coolant flow through the control rod channel. The joint made between the absorber section and the follower has been designed to allow maximum handling flexibility in that a single handling tool is used to handle either the absorber or the follower. In addition, since the joint can be "broken," less head room is needed during handling operations.

In the reactor core, the types of fuel assemblies are limited to two. This simplifies the fabrication process and loading of the core, and results in a core having 32 cruciform slots of which only 24 are occupied by movable control rods. The remaining eight cruciform slots are filled by fixed shim rods consisting of a boron stainless steel section and a Zircaloy-2 section. Each end of the rod is mechanically identical with the other, so that the rod may be inverted and either section may be used as material in the core, as required for proper reactivity and/or flux distribution.

Core Structure

The Yankee reactor core structure consists, in general, of an upper and lower core support plate, an upper and lower core

support barrel, a core barrel and a baffle structure. The major functions of the core structure are to:

(1) Support the weight and maintain the orientation of the fuel assemblies.

(2) Support and secure the position of the control rod shroud tubes.

(3) Absorb the impact of the control rods on the upper core support plate during a scram.

(4) Maintain control rod orientation.

Each core support plate is a rigid assembly of two perforated plates. The upper and lower plates of each support plate are 1.25 and 1.625 in. thick, respectively. The plates are joined by 76 sleeves welded directly to the plates and which position the fuel assembly handling sockets. The sleeves, together with a spacer ring on the outer edge of the plates, stiffen the two plates and form a rigid sandwich type support structure. In addition to supporting the fuel assemblies, the support plates position the control rods by means of guide blocks bolted and dowelled to the support plates. The control rod shrouds are positioned by bolts to the lower core support plates. Dashpot stops are bolted securely to the upper face of the upper core support plate.

The baffle structure separates the incoming cooling water flowing downward outside of the core from that flowing upward through the core. Reinforcing ribs, running axially at strategic positions along the baffle walls, strengthen the structure to withstand the hydraulic pressure differential. As a secondary function, the baffle structure, of 0.5 in. thickness, acts as an effective attenuator for gamma rays from the core.

The core barrel, fabricated from 1.0 in. plate, is located between the baffle structure and the thermal shield. The lower core support plate is dowelled and bolted to the lower rim of this barrel. The upper rim of the core barrel and the upper flange of the baffle are dowelled and bolted to the lower flange of the lower core support barrel. The upper flange of the lower core support barrel is supported on a circumferential ledge machined in the pressure vessel wall at the level of the reactor vessel flange.

The upper core support plate is dowelled and bolted to the lower rim of the upper core support barrel. The top flange of the upper core support barrel rests upon the upper flange of the lower core support barrel. The guide tube support plate is supported and positioned by the upper flange of the upper core support barrel and, in turn, supports the guide tube hold-down plate and ring. Fig. 2 shows the arrangement of these structures.

The entire core support structure rests as an integral unit on the support ledge of the vessel and is retained by a compression load that is applied to the core hold-down ring by the vessel head. The core hold-down ring acts as a large Belleville spring with an induced compression load designed to have a greater force than that which results from the pressure differential across the core. Four guide pins orient the lower core support barrel with respect to the vessel, and four additional guide pins orient the remaining components which rest upon the upper flange of the lower core support barrel.

Neutron Source

The neutron source consists of both polonium-beryllium sources and regenerative antimony-beryllium sources. The polonium-beryllium sources are of sufficient strength for the initial loading, start-up, and testing of the reactor. The antimony-beryllium sources will become activated during operation of the reactor as the polonium-beryllium sources decay. All sources are contained in stainless steel tubes mounted on four vertical vanes each extending a few inches into a different quadrant of the core. Fig. 7 shows the position of these vanes.

Thermal and Hydraulic Design

The thermal and hydraulic design of the initial reactor core has been developed by Westinghouse on the basis of the following conditions:

(1) Steam conditions at 392 MW at the outlet of the steam generator are 525 p.s.i.g., 475°F; the log mean temperature difference in the steam generator is 37°F at full load.

(2) The maximum heat flux in the core is not to exceed 50% of the burn-out heat flux as predicted by the Bettis Correlations[3].

(3) Local boiling, or surface boiling of the sub-cooled liquid, is permissible within the core.

(4) Bulk boiling is not permitted within the core under normal operating conditions.

In addition to the above physical limitations, the maximum power output of a nuclear reactor is limited by the ratio of maximum to average power density within the core. This ratio is arrived at by considering both engineering and nuclear effects and is expressed as a "hot channel factor." The engineering effects are basically statistical in nature and are due to non-uniformities in fuel rod characteristics and flow distribution. The nuclear factors affecting the maximum-to-average power density are due to variations in neutron flux within the core.

The resultant design data are given in Table 3.

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1. Lessman, G. R. "End Closures for the Yankee First Core Fuel Rod," YAEC–110, May, 1959.
2. King, P. P. "Joining Fuel Rods into Sub-assemblies for YAEC Fuel Elements," YAEC–108.
3. De Bortoli, R. A. *et al.* "Forced Convection Heat Transfer Burnout Studies for Water in Rectangular Channels and Round Tubes at Pressures Above 500 p.s.i.a." WAPD–188.

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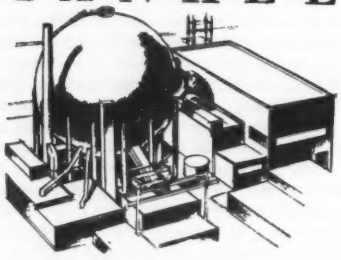


Control Rod Drive Mechanism

by THOMAS F. WIDMER

(*Supervisor of Mechanical Design, Westinghouse Atomic Power Department*)

CONTROL rods in the Yankee core are operated by a set of 24 drive mechanisms mounted on the vessel head. These mechanisms are all identical, and fulfil the functions of regulation, shim and safety for each rod. Design of the mechanisms is based upon the magnetic jack principle which provides the advantage of hermetic closure for the primary system. This concept was chosen because of the high pressure (2,500 p.s.i.a.) and stringent leakage requirements which limit the use of mechanisms having shaft seals or packing. Any mechanically sealed penetration of the reactor head is undesirable, not only from the leakage standpoint, but also because of the differential pressure forces tending to eject the seal shaft or rod from the reactor.

As the Yankee mechanisms were restricted to a hermetically sealed design, the only mechanism types which appeared practical were the canned motor-roller nut type and the relatively newer magnetic jack. The weight of Yankee rods (400 lb) indicated that wear might limit the life expectancy of canned roller nut type mechanisms, and therefore, development of a magnetic jack was undertaken. Specific objectives considered in the Yankee design were as follow:—

(1) Operating life of about 2,000 full excursions.

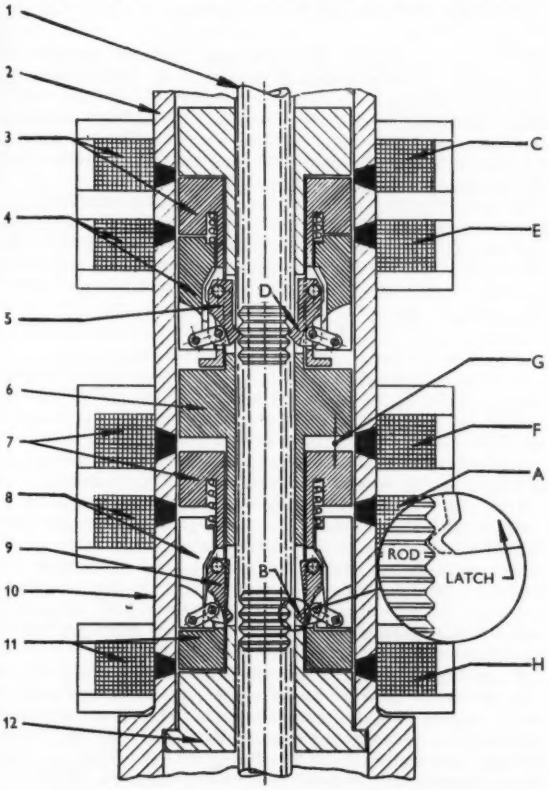
(2) Lifting capacity of at least 750 lb.
(3) Rod positioning rate up to at least 10 in./min.
(4) Reliable scram up to a friction coefficient of 1.0.
(5) Maximum mechanism diameter of 10.5 in.
(6) Position of control rod continuously indicated.
(7) Pressure shell designed in accordance with ASME Code, Section VIII.
(8) Seal welded closure between mechanism and reactor vessel head.
(9) Only air cooling for electrical coils.
(10) Electrical coils removable without disassembly of the mechanism or detachment from the reactor vessel head.

The Yankee mechanism which has been developed to meet these requirements is shown in Fig. 8. Details of construction and operation are best illustrated by Fig. 9, a schematic of the mechanism's working parts. All mechanical operations take place inside the pressure containment shell, and no moving parts penetrate this boundary.

Motion of the rod is accomplished by sequential engagement of the two latch assemblies "B" and "D" of Fig. 9. Force is applied to the latches by means of magnetic flux driven through the pressure shell from five solenoid coils (A, C, E, F and H) surrounding the shell. The exact sequence of operation in a typical lift cycle is described in the accompanying scheme which will serve to illustrate how the long life objective is met by eliminating the major cause of latch tooth wear.

Repetition of this cycle, at frequencies up to 40 steps per minute in either direction, allows the rod to be moved at an average rate of 15 in./min maximum. Power is supplied to the mechanism by a series of cam operated switches which actuate magnetic contactors feeding direct current pulses to the coils to produce the steps described above. Prototype life tests under simulated reactor conditions have demonstrated successful operation with only nominal wear through the design life of 2,000 excursions. This is equivalent to about one million individual steps of $\frac{1}{8}$ in.

Fig. 9. Yankee control mechanism schematic.
1. Drive rod. 2. Load transfer magnet pole. 3. Load transfer coil and armature. 4. Stationary gripper and armature. 5. Stationary gripper latches. 6. Lift coil magnet pole. 7. Lift coil and armature. 8. Movable gripper coil and armature. 9. Movable gripper latches. 10. Pressure housing. 11. Pull down coil and armature. 12. Pull down magnet pole.



Two distinct advantages are inherent in the step notion of this mechanism. No torque is applied to the control rod to cause wear, and the digital nature of the cycle precludes any possible overspeed during rod withdrawal despite errors in the command signal, because the coil inductances limit the maximum cycling rate. Wear of latch arms and drive shaft teeth has been virtually eliminated by the load transfer feature of the design. Control of side thrust forces on magnet armatures through accurate centring, and introduction of fixed radial flux gaps has also largely eliminated wear on the chromium plated sliding surfaces which guide these armatures.

Reliable scram is assured by conservative selection of tooth and linkage geometry in the latch assemblies. Yankee geometry allows scram to occur at friction coefficients as high as 1.55. Hot tests indicate that scram motion of the drive shaft begins within 0.15 sec of power interruption to the mechanism latch coil.

Further, a spring is provided at each latch magnet armature to assure quick return of the latch mechanism during normal cycling, but these springs produce only 30 lb force and are not required for scram. The mechanism drive shaft assembly is completely independent of the mechanism when all coils are de-energized. This allows easy removal of the complete reactor vessel head and mechanisms during refuelling. Each control rod drive shaft attaches to one of the cruciform shaped control rods by means of a latch. An unlocking rod allows the control rod to be disconnected from the drive shaft by means of a remote handling tool which grasps the uppermost end of the shaft. Each drive shaft assembly carries a piston and cylinder type hydraulic shock abosrber to limit the decelerating forces on the control rod and core plate structure caused by scramming.

Based upon the 10.5 in. diameter limitation, the various parts were proportioned to fit this available space. Using a 1.5 in. diameter drive shaft, the required magnet pole area to produce 750 lb lifting force established a pressure shell of 4.75 in. i.d. The necessity of selecting a corrosion resistant magnet armature

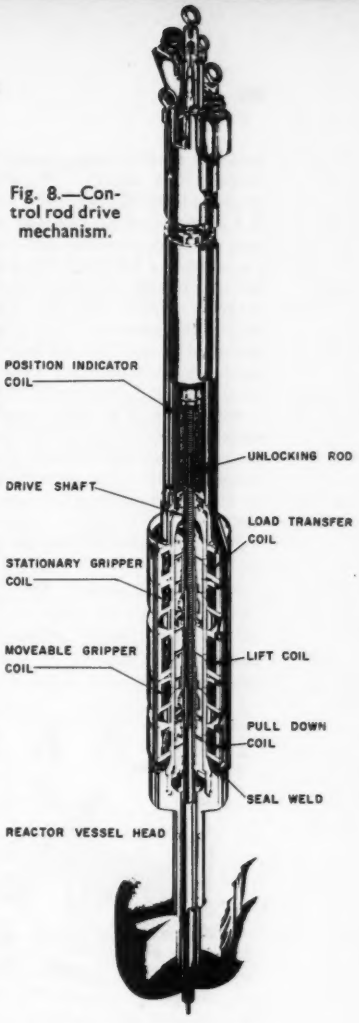


Fig. 8.—Control rod drive mechanism.

1. Assuming that the cycle begins with stationary gripper energized, movable gripper magnet (A) is excited so that latch arm (B) is engaged in a drive shaft notch. Note that a 1/32 in. axial clearance exists between surface of latch tooth and corresponding shaft tooth.
2. Load transfer magnet coil (C) is de-energized, allowing the stationary gripper assembly and the drive shaft to drop until clearance is taken up at the movable gripper tooth (B). Stationary gripper assembly continues to drop an additional 1/32 in., thereby completely removing load from the stationary gripper latch arm (D).
3. Stationary gripper magnet (E) is de-energized, allowing latch arm (D) to swing out of engagement.
4. Lift coil (F) is energized to close the $\frac{1}{8}$ in. gap (G), thereby lifting the movable gripper and driveshaft through one step.
5. Stationary gripper magnet (E) is energized, allowing latch arm (D) to enter a notch in the driveshaft. Note again that the tooth surface is not loaded at this time because there is axial clearance between the latch and rod tooth.
6. Load transfer magnet coil (C) is energized to lift the stationary gripper assembly into contact with the driveshaft. After clearance is taken up, the rod lifts an additional 1/32 in. to remove load from the movable gripper teeth.
7. Movable gripper magnet (A) is de-energized, permitting latch arm (B) to swing clear.
8. Lift coil (F) is de-energized and pull down coil (H) is energized to return the movable gripper in preparation for the next stroke.

material (410 stainless steel) penalized the magnet design by limiting flux density. Annealed 410 stainless steel saturates at about 120 k Maxwell/in.^{2}, so a design point of 70 k Maxwell/in.^{2} at the working gap was used to arrive at 12.6 in.^{2} of pole face, thereby allowing sufficient margin for leakage flux without excessive demands on coil excitation. Total output of the lift coil is approximately 10,000 ampere turns. Latch locking magnets require less pole area than the lift magnet and fit within the same 4.75 in. pressure shell. These magnets are designed to hold the latches in engagement under impact load (about 5 g) encountered at the end of each $\frac{1}{8}$ in. rod step. An excitation of 3,500 ampere turns is used for each gripper coil.

With the 4.75 in. inside diameter, a wall thickness of approximately 0.625 in. is required to satisfy ASME Code allowable stresses, thus establishing a 6 in. outside diameter for the pressure shell. The remaining space, 2.25 in. per side, is taken by electrical coils and flux conductors to complete each magnetic circuit.

Rod position information is discussed in the article which follows this.

As the mechanism operation depends upon magnet flux forced through the walls of a thick pressure shell, the design of this shell strongly influences the efficiency of the magnetic circuits. Reluctance of the circuits is minimized by using permeable material (AISI 405 stainless steel) for the portion of the pressure shell surrounding the mechanism working parts. At each coil, however, it is necessary to insert a circumferential band of non-permeable material (AISI 308 stainless steel) to prevent short circuiting of the magnet flux. The composite housing required in this design is made by depositing AISI 308 weld material into grooves cut around a tube of 405 stainless steel. After welding, the tube is bored and turned to finish dimensions so that weld root passes are machined out, leaving alternate rings of AISI 405 and 308 stainless steel. Non-permeable 308 material appears as a dark band adjacent to each coil in Fig 9. The complete pressure shell screws into a port in the reactor head, and is seal welded at that point.

All five operating coils and their flux conductors are assembled as a removable stack which slides over the pressure shell. These coils are wound with glass insulated copper wire, and are impregnated with a high temperature silicone resin to reduce copper oxidation and improve heat transfer. Tests have shown that coil electrical losses and heat transfer from the reactor can readily be absorbed by forced air circulation. Coil temperatures during continuous operation do not exceed 200°C as compared with 250°C, the safe limit of their insulation.

A more recent version of the magnetic jack has been designed for control rods weighing up to 1,000 lb. In preliminary tests, this mechanism has demonstrated a static lifting capacity of 2,300 lb, yet requires no more space than the 10.8-in. pitch of the Yankee mechanism.

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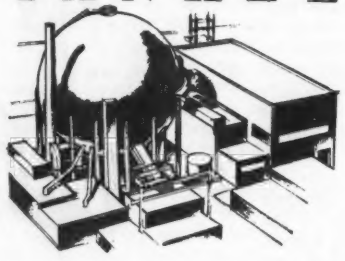


Reactor Control System

by CLOVIS F. OBERMESSER

(Manager, Systems Analysis and Control Plant Development)

THE Yankee reactor control system as designed by Westinghouse provides for manual or automatic control of the reactor, control switching for the 24 latch type rod mechanisms, and two separate means of rod position indication.

The system equipment consists of:—

(1) Reactor Servo Control.
(2) Control Rod Drive Mechanisms.
(3) Rod Position Indications.
  (a) Primary Rod Position Indication.
  (b) Secondary Rod Position Indication.

Reactor Servo Control

The servo control portion of the system provides manual or automatic reactor control for an essentially constant average temperature maintained on a steady state basis. It is capable of restoring system average temperature following a scheduled or transient change in load, and can also compensate for reactivity changes due to fuel depletion and/or xenon level changes. It will initiate rod motion automatically at power levels greater than 10% of full power, and provide a means for manually initiating rod motion at any power level.

The principal equipment, shown in Fig. 10, is mounted on the nuclear control board and consists of:—

(1) Eight toggle switches; one for each of the main loop temperature legs; four hot and four cold.
(2) Two auctioneering units; one hot, one cold, for selecting the highest temperature in each.
(3) An average temperature computer; averages the auctioneers.
(4) An average temperature indicator.
(5) A temperature summing unit with calibrated adjustable reference source combines T_{av} and T_{ref}.
(6) Bi-stable magnetic amplifiers for: rods-in, rods-out, fail-safe.

The characteristic temperatures are shown in Table 4. The T_{av} signal is combined with the T_{ref} signal (normally set for 514°F) and transmitted to a difference indicator mounted on the control board. Provision is made to reduce the T_{ref} signal by an adjustable step in the range of −3°F to −10°F. When the absolute difference between T_{ref} and T_{av} exceeds a manually adjustable value of from 3 to 30°F, the magnetic amplifier causes a "fail-safe" alarm to sound, thereby notifying the operator of failure of T_{av} or T_{ref}.

If the difference (T_{av}—T_{ref}) exceeds +3°F (TRIP-ON), a magnetic amplifier initiates "IN" rod motion and discontinues rod motion only when the sum reaches +2.5°F (TRIP-OFF). TRIP-ON, TRIP-OFF, and loop width are manually adjustable. Similar circuit design applies for "OUT" rod motion.

Control Rod Drives

Energizing the same functional coils of more than one control rod mechanism simultaneously allows these mechanisms to be controlled as a group. The 24 control mechanisms have, in fact, been grouped in the following manner: three groups of four, two groups of two, and one group of eight. A partial system block diagram of this arrangement is shown in Fig. 11.

Two auxiliary inversion relays are provided to arrange the rod group position indicating circuits for either "Rods In" or "Rods Out" indication, depending upon the respective signal. "All Rods In" motion is available at all times and takes precedence over any motion already in progress. This motion is accomplished by merely switching to the "All Rods In" position. Should it become necessary to operate any group with less than the normal number of rods, the pull down and lift coils of any mechanism which is to remain stationary can be switch disconnected.

Removing all power to the mechanisms permits the control rods to drop. In order to scram, therefore, the trip coils on power supply breakers are energized. This is accomplished manually by pressing any one of the three "Scram" push-buttons; and automatically by closing the relay contacts which are operated from associated control and monitoring systems.

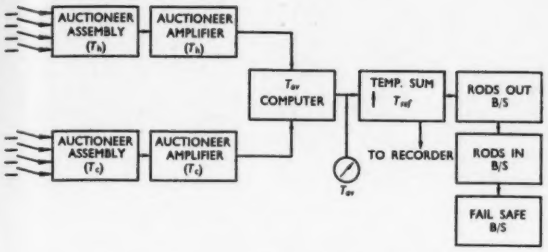
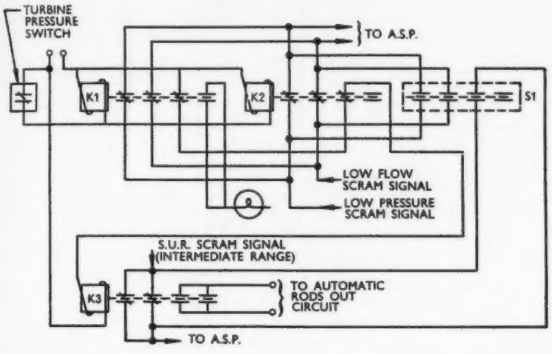


Fig. 10 (above).—Reactor Servo control. Fig. 12 (below).—Permissive relay circuit.



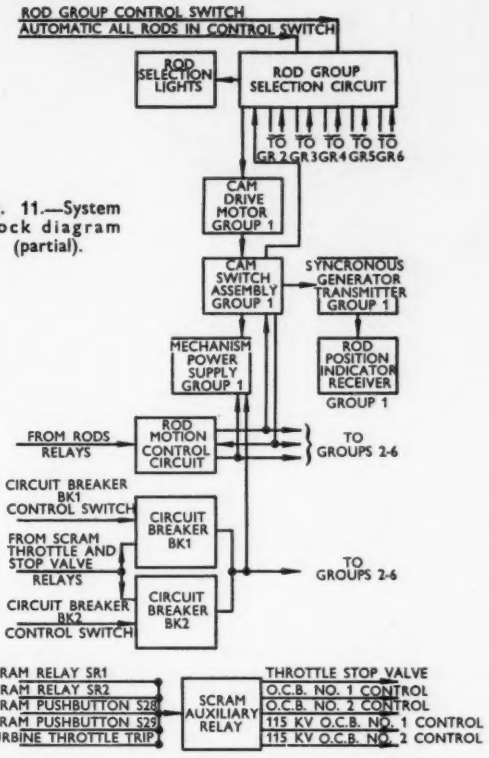


Fig. 11.—System block diagram (partial).

Rod Position Indication

The primary rod position indication scheme consists of 30 3-in. transformers per control rod mechanically connected to facilitate handling, and mounted around the pressure housing which contains the control rod drive shaft. The primary windings of five transformers are in series and maintain an essentially constant primary current. A 2-V bulb across each secondary winding corresponds to its respective transformer secondary position around the control rod drive shaft. As the control rod is withdrawn from the reactor, the increased permeability of each successive transformer increases the coupling between the primary and secondary winding enough to light the bulbs successively, giving a positive column type indication of each individual control rod position.

A secondary rod position indicating scheme consists of six self-synchronous transmitters mounted on the control rod camshaft extensions in the control rod mechanism power supply, and six self-synchronous receivers mounted on the control board. Six digital read-out indicators are mounted on the control board with provision for manual reset. The unit counts the number of revolutions made by the camshafts, indicating each rod group position with an accuracy of $\pm\frac{1}{8}$ in. (assuming that the rod has not dropped).

Nuclear Instrumentation

Reactor flux monitoring from source to 150% of reactor designed full power output, is provided by the nuclear instrumentation. This equipment incorporates provisions for initiating a reactor and turbine shut-down if conditions exist which may be hazardous for plant operation. Circuitry is also provided to bypass several shutdown signals for start-up operation.

A continuing Westinghouse programme to minimize vacuum tubes in this capacity has resulted in an overall simplification such that there are less than 60 tubes total, including the power supplies in the start-up range instrumentation. In the power range, only magnetic amplifiers are used.

The equipment consists of two identical Source-Ranges, two identical Intermediate-Ranges, three identical Power-Ranges and one special Intermediate-Range. The Source-Range and Intermediate-Range are designated as the Start-up Range.

Reactor neutron flux measurement is performed in the neutron shield tank surrounding the reactor vessel. The detectors, in their container assemblies, are lowered to the appropriate level into thimbles welded to the neutron shield tank. Three of these thimbles, placed approximately 120° from one another, each contain one intermediate and one power range ionization chamber located at different elevations, one above and the other below the centre of the reactor. By checking the changes and relative flux reading shown, tendencies toward the development of flux tilt will be detected.

The source-range detectors (channels Nos. 1 and 2) consist of BF_3 proportional counters. These counters have a sensitivity of approximately 4.5 counts/neutron/cm² sec, and cover a flux range of 2.5×10^{-1} to 2.5×10^{4} nv, corresponding to detector output pulses of 1 count/sec to 100,000 count/sec.

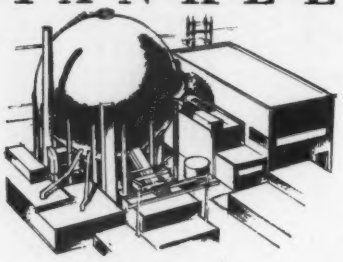
The intermediate-range detectors (channels Nos. 3, 4 and 5) consist of compensated ionization chambers. These chambers have a neutron sensitivity of approximately 4×10^{-14} amp/neutron/cm² sec and a gamma sensitivity of 3×10^{-13} amp/r/h. The compensated ionization chambers cover a flux range of 2.5×10^{2} to 2.5×10^{10} nv. The intermediate-level meter is calibrated from 10^{-11} amp to 10^{-4} amp. An output is also provided which is proportional to the rate of change of the logarithm of the neutron flux level.

Three linear meters are used for flux tilt measurements and are connected in series with the high voltage power supplies and the compensated ionization chambers. These meters provide information over two decades of the power-range. A variable high-voltage output from the power-range channels allows for operation of the compensated ionization chambers in the high flux regions. The power-range detectors (channels Nos. 6, 7 and 8) consist of uncompensated ionization chambers. These chambers have a neutron sensitivity of 4.4×10^{-14} amp/neutron/cm² sec and a gamma sensitivity of 5×10^{-11} amp/r/h. The uncompensated ionization chambers cover a flux range of 2.5×10^{4} to 2.5×10^{10} nv.

TABLE 4. SYSTEM TEMPERATURES

Temperatures	Range, °F
Hot leg, T_h	510–540
Cold leg, T_c	485–515
Reference, T_{ref}	497–527

YANKEE



Fuel Handling

by HARLEY A. SMITH

(Asst. to the Manager, Plant Development Auxiliary Systems, Westinghouse Atomic Power Department)

FUEL-HANDLING facilities at Yankee shown in Fig. 13 include means for underwater removal of spent fuel assemblies and control rods from the reactor, and a system for transferring such spent assemblies from inside the vapour container to the external storage pit. Here the spent fuel is stored under water and allowed to decay to a tolerable level before subsequent loading into a shipping container, removal from the plant site, and shipment to a fuel reprocessing facility. New fuel, as it is fabricated and shipped to the plant site, is stored in a dry vault adjacent to the spent-fuel pit.

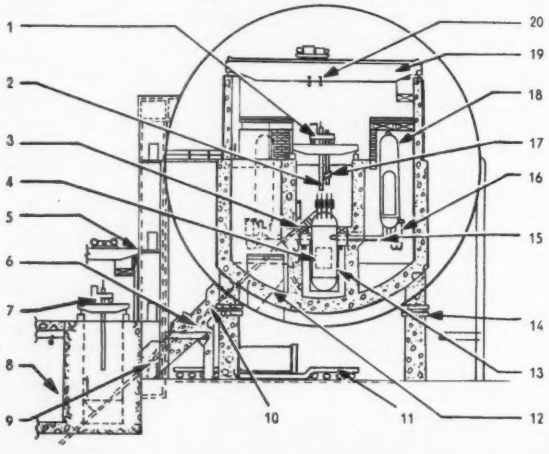
Four parallel rows of racks in the storage vault will hold 84 new fuel assemblies, 24 control rods and eight shim rods. These racks are arranged so as to maintain a surface-to-surface spacing of 6.3 in. between assemblies. This arrangement ensures that the fuel array is always in a safe status, even during a maximum credible accident such as vault flooding. Calculations show that even if the vault were to be flooded, k_{eff} would be approximately 0.8.

Fuel assemblies destined to be loaded in the reactor are removed from the storage rack and transported to a point directly under a hatchway in the vault roof through which they are lifted for transfer to the new fuel elevator through the pit hatch. This elevator, in turn, lowers the new fuel assemblies into the proper position for pick-up by the spent fuel transfer pit manipulator crane.

The geometry of the spent fuel racks prevents any possible occurrence of criticality. This has been accomplished in a minimum area by an alternate spacing array, which maintains a centre-to-centre distance of 19½ in. and approximately 12 in. surface-to-surface distance between spent-fuel assemblies. The rack layout is designed to prevent insertion of fuel assemblies into any locations except those that are exactly prescribed. Decay heat emitted by the spent-fuel assemblies is removed from the fuel-transfer pit water by a 500-gal/min circulating pump and a heat exchanger which maintains the water temperature below 130°F. This same pump also circulates the pit water through a 20-ft³ ion exchanger, thereby preventing any build-up of radio-activity and boric acid in the pit water. As an additional function, the pump removes the water from the pit when necessary.

Fig. 13.—Fuel handling arrangements.

1. Shield tank cavity manipulator crane. 2. Tool boom and UHT. 3. Upper lock valve. 4. Reactor. 5. 75-t crane. 6. De-watering system. 7. Spent fuel pit manipulating crane. 8. New fuel storage vault. 9. Lower lock valve. 10. Spent fuel chute. 11. Shipping car. 12. Shipping hatch. 13. Shield tank. 14. Plate seal. 15. Main coolant loop. 16. Primary pump. 17. T.V. boom. 18. Steam generator. 19. 150-t polar crane. 20. Two 75-t hooks.



Fuel-transfer Chute and Carriage

The safe continuity of underwater transfer of fuel and control rods is maintained by an inclined transfer chute between the shield tank cavity and the spent-fuel transfer pit. This is accomplished with a minimum-size penetration through the vapour-container wall. The chute consists of sections of 12-in. i.d. stainless-steel pipe flanged together, a motor-operated lock valve near the lower end of the chute, and a swinging plate lock valve at the upper end. Near the upper end of the chute, just outside the vapour container, there is a flanged joint with provision for inserting a solid plate to maintain vapour-container integrity during plant operation. Before refuelling begins, a plate with a 12-in. i.d. hole is substituted for the solid plate, thereby allowing passage of the transfer carriage during refuelling. Tracks for supporting and guiding the transfer carriage extend the full length of the chute. A hinged joint in the carriage permits the receptacle end of the carriage to be raised to the vertical position by the laydown devices. Co-ordination is thereby achieved with the vertical transfer functions of the manipulator cranes.

The upper plate lock-valve is spring loaded in the closed position. It opens upon contact with the carriage, and closes after the carriage has passed through the valve on its downward travel. The lower lock-valve is motor operated and is controlled by the operator on the shield tank cavity manipulator crane. This valve remains closed until the carriage travel indicator lights show that the transfer carriage has descended to a point just above the valve. At this stage, in order to minimize the loss of borated water from the shield tank cavity to the transfer pit, the mixture of borated water and demineralized pit water is pumped out of the chute and back to the shield tank cavity. This pumping is continued until the chute water level recedes to a level corresponding to the pit water level. Elevations are arranged to ensure complete water coverage of the spent fuel assembly at all times during the transfer and de-watering process.

After the indicator light shows that the de-watering is completed, the operator opens the lower lock-valve, allowing the transfer carriage to descend to the lowest extent of its travel into the transfer pit. The receptacle portion of the carriage is then raised to the vertical position by the laydown device and the spent fuel assembly is lifted out by the fuel transfer pit manipulator crane. The process is continued until the fuelling or refuelling operation is complete.

During refuelling operations, the reinforced concrete stainless steel lined shield tank cavity is filled to a depth of approximately 25 ft with borated water, which also serves as a cooling medium for the spent fuel. In addition, this cavity provides a convenient laydown and storage area for the reactor vessel head with its lifting device, and for the upper reactor vessel internals. Radiation at the surface of the water is limited to a level of 2 mr/h by a minimum depth of 13 ft of water shield over a withdrawn fuel assembly, and 8 ft minimum over a withdrawn control rod-follower assembly.

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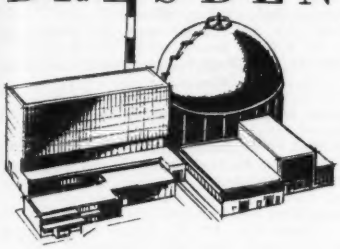

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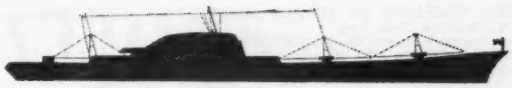
CHICKSAN CO., 600 16th Street, Oakland, California.
CLARK EQUIPMENT CO., c/o Lifton Incorporated, 6601 San Leandro Street, Oakland 21, California.
J. B. CLOW AND SONS, 201-299 N. Talman Avenue, Chicago 80, Illinois.
COAST EQUIPMENT CO., 444 8th Street, San Francisco 1, California.
COMBUSTION ENGINEERING, 155 Sansome Street, San Francisco 4.
CONCO ENG.-WRS. (Division of H. D. Conkey and Co.), Mendots, Illinois.
CONTINENT CONCRETE PIPE CO., 285 N. LaSalle Street, Chicago 1.
G. A. COOKE CO., 920 Grayson Street, Berkeley 10, California.
CRAEMER AND DUNLOP INC., West 21st Street, W. of Union Street, Tulsa 1, Oklahoma.
CRANE CO., 301 Brannan Street, San Francisco, California.
CROSBY VALVE AND GAGE CO., 16 Beale Street, San Francisco, California.
CYCLOPS, 837-847 Folsom Street, San Francisco, California.
WM. M. DEE CO., 4808 West Wilson Avenue, Chicago 30, Illinois.
DELAVAL TURB. PAC. CO., 201 East Millbrae Avenue, Millbrae, California.
DOLLINGER, c/o Strout Equipment Co., 630 Natona Street, San Francisco.
DUPRIZIO CONSTRUCTION CO., 3001 Saffell Avenue, Melrose Park.
DYER CONSTRUCTION CO., 1716 Sheffield Avenue, Dyer, Indiana.
MOESCHL EDWARDS CO., P.O. Box 1115, Cincinnati 1, Ohio.
ELECTRO RUST-PROOFING, 2180 Palou Avenue, San Francisco, California.
ENDURANCE METAL PRODUCTS CO., 146 South Maple Avenue, South San Francisco, California.
FABRI-VALVE.
FAIRBANKS-MORSE CO., 630 3rd Street, San Francisco 7, California.
FISCHER AND PORTER, 5010 Woodminster Lane, Oakland 2, California.
WALTER H. FLOOD AND CO., 6102 South Blackstone Avenue, Chicago 37.
FLORE PIPE CO., 601 E. Red Bud Avenue, St. Louis 15, Missouri.
L. B. FOSTER CO., 231 S. LaSalle Street, Chicago, Illinois.
GARLOCK PACKING CO., 780 18th Street, San Francisco 7, California.
GENERAL ELECTRIC, 235 Montgomery Street, San Francisco 6, California.
G. E. SUPPLY CO., 1201 Bryant Street, San Francisco, California.
GILMORE STEEL AND SUPPLY, 840 Brannan Street, San Francisco.
GRAVER TANK AND MFG. CO., 155 Montgomery Street, San Francisco 4.
GRAYBAR ELECTRIC CO., INC., 6415 Indianapolis Bend, Hammond, Indiana.
CLOW GRINNELL COLUMBIA PIPE CO., 5th and Brannan Street, San Francisco, California.
GRINNELL, 425 South Western Avenue, Chicago, Illinois.
GRISCOM-RUSSEL CO., 55 New Montgomery Street, San Francisco.
D. W. HAERING CO., P.O. Box 6037, San Antonio, Texas.
HILLS McCANNA, c/o Stiehl and Co., 605 Addison Street, Berkeley 10.
HOLBROOK MERRILL CO., 1000 25th Street, San Francisco, California.
A. C. HORN, 1215 W. Washington Blvd., Chicago, Illinois.
E. M. HUNDLEY HARDWARE CO., 662 Mission Street, San Francisco 5.
RODNEY HUNT MACHINE CO., 7 Water Street, Orange, Massachusetts.
ILLINOIS BELL TELEPHONE CO., 131 North Franklin Street, Chicago 6.
ILLINOIS CORR. CULVERT CO., Peoria, Illinois.
IMPERIAL STEEL TANK CO., 3234 W. 31st Street, Chicago, Illinois.
INGERSOLL-RAND, 350 Brannan Street, San Francisco 7, California.
INLAND SUPPLY CO., 21 Osgood Street, Joliet, Illinois.
INTERNATIONAL IRON WORKS, 3120 South Austin Blvd., Cicero 50.
C. B. ISETT AND SON, INC., 3035-43 N. Rockwell Street, Chicago 18, Illinois.
JACOBS ENG. CO., 600 16th Street, Oakland, California.
JACUZZI BROS., 5327 Jacuzzi Avenue, Richmond, California.
JOY MFG. CO., 1155 Harrison Street, San Francisco 3, California.
G. M. KEMP CO., c/o E. A. Wilcox Co., 277 7th Street, San Francisco.
KEWAUNEE MFG. CO., 2442 Park Blvd., Palo Alto, California.
LABORATORY EQUIPMENT CO., 649 Bryant Street, San Francisco 7.
LAPP INSULATOR CO., 420 Market Street, San Francisco, California.
LAYNE AND BOWLES, c/o George E. Swett and Co., 100 Howard Street, San Francisco 5, California.
LEEDS AND NORTHRUP CO., 1095 Market Street, San Francisco 3.
LEWISTOWN PIPE CO., P.O. Box 391, La Grange, Illinois.
LINK BELT, 400 Paul Avenue, San Francisco 24, California.
LOUDEN, c/o Acme Equipment Co., 801 Minnesota Street, San Francisco 7.
LYONS METAL PRODUCTS CO., Enterprise Commercial Co., P.O. Box 3006, Rincon Annex, San Francisco, California.
MACFARLAND ENG., 1206 Nance Street, Houston 2, Texas.
R. C. MAHON, c/o Olympic Steel Products, 821 Market Street, San Francisco.
MANNING, MAXWELL AND MOORE, 450 Bayshore Blvd., San Francisco.
J. L. MANTA, INC., 7525 South Chicago, Chicago, Illinois.
JOHNS MANVILLE SALES CORP., 116 New Montgomery Street, San Francisco, California.
MATERIAL SERVICE CORP., 300 West Washington Street, Chicago, Illinois.
MIDWEST PIPE ORDER, 420 Market Street, San Francisco, California.
J. B. MILLIS, 1955 E. 73rd Place, Chicago 49, Illinois.
MINE SAFETY APPLIANCE CO., 201 N. Braddock Avenue, Pittsburgh 8.
MODERN BUILDERS CORP., P.O. Box 1233, Joliet, Illinois.
MOFFET, P.O. Box 54, Albany, California.
MOORE, 7th and Carleton Streets, Berkeley 10, California.
SAMUEL MOORE AND CO., 420 Market Street, San Francisco 11, California.
NATIONAL ALUMINATE CORP., 6216 West 66th Place, Chicago 38, Illinois.
NATIONAL TANK AND BOILER CO., 1822 Cherry Street, St. Louis 20.
NEENAH FOUNDRY CO., 505 Winneconne Avenue, Neenah, Wisconsin.
NEWKIRK TRANSIT MIX, 720 N. Collins Street, Joliet, Illinois.
NOOTER CORP., 1400 South 2nd Street, St. Louis, Missouri.
OTIS ELEVATOR CO., 600 W. Jackson Blvd., Chicago 6, Illinois.
OVERLY MFG. CO., Overly Street, Greensburg, Pennsylvania.
PACIFIC, CHAPMAN GRINNEL, 5th and Brannan Street, San Francisco
PACIFIC PIPE CO., 401 Folsom Street, San Francisco, California.
PACIFIC SCIENTIFIC CO., 25 Stillman Street, San Francisco 7, California.
PENN. PUMP AND COMPRESSOR, 114 14th Street, San Francisco 3
PEERLESS PUMPS, 461 Market Street, San Francisco, California.
PERAUD ROOFING CO., 557 N. Chicago Street, Joliet, Illinois.
IRA G. PERIN CO., 575 Howard Street, San Francisco, California.
(PHELPS DODGE COPPER PROD. CORP.) AND ANDACONDA, 120 Montgomery Street, San Francisco, California.
PIONEER FIREPROOF DOOR CORP., 811 S. Fulton Avenue, Mount Vernon, New York.
POWELL, 461 Market Street, San Francisco, California.
HENRY PRATT, 1639 Russ Building, San Francisco 4, California.
PUTNAM ORG., North Pier Terminal Bldg., 539 E. Illinois Street, Chicago 11.
PYRENE (CO_2 Division), 140 Hawthorne, San Francisco, California.
R.C.A., 90 Tehama Street, San Francisco 5, California.
REPUBLIC SUPPLY CO., 1919 Williams Street, San Leandro, California.
RICHARDS-WILCOX, 850 South Van Ness Avenue, San Francisco 10.
ROCK ISLAND BRIDGE AND IRON CO., 1603 Mill Street, Rock Island, Illinois.
RUST ENG., 930 Fort Duquesne Blvd., Pittsburgh 22, Pennsylvania.
RYERSON, P.O. Box 8000-A, Chicago 80, Illinois.
SANFORD MECHANICAL EQUIPMENT, 5658 College Avenue, Oakland, California.
SERVICE ENGINEERS, 3030 West Pafford, Fort Worth 9, Texas.
SIER BATH, 114 14th Street, San Francisco, California.
SIMMONDS MACHINERY CO., 816 Folsom Street, San Francisco, California.
SOLAR AIRCRAFT CO., 16 Beale Street, San Francisco 5, California.
SOUTHWESTERN ENG. CO., 4800 Santa Fe Avenue, Los Angeles 58, California.
SPRINGFIELD BOILER CO., c/o Far Western Eng. and Chem. Co., Bldg. 631, Oakland Airport, Oakland, California.
STAPLES AND PFEIFFER, 528 Bryant Street, San Francisco, California.
A.M. STEEL AND WIRE CO. Div. U.S. Steel Corp., 440 Bayshore Blvd., San Francisco, California.
TAYLOR FORGE, P.O. Box 485, Chicago 90, Illinois.
TRANE, 1485 Bayshore Blvd., San Francisco 24, California.
TRIDENT ENGINEERING CO., 16 Beale Street, San Francisco 5, California.
VELAN ENGINEERING, 45 California Street, San Francisco, California.
WALLACE AND TIERNAN, 2180 Palow Avenue, San Francisco, California.
WALWORTH CO., 155 Montgomery Street, San Francisco, California.
WARREN PUMPS, c/o Geo. E. Swett and Co., 100 Howard Street, San Francisco.
L.A. WATER SOFTENER CO., c/o R. F. MacDonald Co., 1485 Bayshore Blvd., San Francisco 24, California.
WEHLING WELL WORKS, 615 Reed Street, Beecker, Illinois.
WERDEN BUCK CO. AND CONTINENTAL CONCRETE PIPE CO., 55 East Webster Street, Joliet, Illinois.
WESTCO PUMP SALES, 85 Industrial Street, San Francisco, California.
WHITING, P.O. Box 307, Norwalk, California.
WILDMAN BOILER AND TANK CO., 3026 Carroll Avenue, Chicago 12.
WILLIAMS AND LANE, 1077 Eastshore Highway, Berkeley 10, California.
WORTHINGTON, 224 Townsend Street, San Francisco 7, California.
WRIGHT HOIST (Div. American Chain and Cable), 890 Tennessee Street, San Francisco, California.

SAVANNAH Contractors

Main Contractors

New York Shipbuilding Corporation

Babcock & Wilcox



MAJOR VENDORS TO NEW YORK SHIPBUILDING CORPORATION:

ALLIS CHALMERS MFG. CO., 864 S. 70th St., Milwaukee, Wisconsin.
Pumps, fire and bilge; transformers.

AMERICAN ENGINEERING, Armingo and Serpviva Sts., Philadelphia, Pa.
Steering gear.

AMER. LAUNDRY AND MACHINERY CO., 40 E. 34th St., New York City, N.Y.
Laundry equipment.

AMER. METAL BEARINGS, 16th and Imperial St., Los Angeles, Calif.
Shaft bearings.

ARNOT-JAMESTOWN DIV. AETNA STEEL PRODUCTS, 730 Fifth Ave., New York 19, N.Y.
Joiner work and furniture.

ATWOOD AND MORRILL CO., Salem, Mass.
6 in. orifice valve.

AURORA PUMP DIVISION OF NEW YORK AIR BRAKE CO., 4610 Loucks St., Aurora, Ill.
Bilge and ballast, fire, salt water, air conditioning, and sump pumps.

AUTH ELECTRIC CO., INC., 34-20 45th St., Long Island City, N.Y.
Wheelhouse alarm panel; hospital and stewards call systems; burglar alarm equipment.

AUTOMATIC SWITCH CO., Florham Park, New Jersey.
Auto. bus transfer equipment.

AVONDALE MARINE WAYS, Box 1030, New Orleans, La.
Steel castings.

BAILEY REFRIGERATION CO., 115 King St., Brooklyn 31, N.Y.
Refrigerators.

BALDT ANCHOR AND CHAIN DIV. OF THE BOSTON METAL CO., P.O. Box 350, Chester, Pa.
Anchors and chains.

BESLER CORP., Ft. of Harlan Ave., Oakland 8, Calif.
Emergency boilers and pumps.

BETHLEHEM STEEL CO., Suburban Station Bldg., Philadelphia, Pa.
Propeller and steel plates.

JAMES BIDDLE CO., 1312 Arch St., Philadelphia, Pa.
Portable electric instruments.

BIRDSBORO STEEL FOUNDRY AND MACHINE COMPANY, 49 Furnace Street, Birdsboro, Pa.
Steel castings.

G. S. BLAKESLEE AND CO., 1822 S. Laramie Ave., Chicago 50, Ill.
Galley equipment.

BLUDWORTH MARINE DIV. OF KEARFOTT CO., INC., 1378 Main Ave., Clifton, N.J.
Echo depth sounder.

BOSTON AND LOCKPORT BLOCK CO., 110 Condor St., Boston, Mass.
Rigging gear.

BROWNELL ASSOCIATES, INC., 135 E. 42nd St., New York 17, N.Y.
Cooling coils.

BUFFALO FORGE CO., 15th and H Sts., Washington 5, D.C.
Ventilation fans.

A. M. BYERS AND CO., Farnam Clark Bldg., Pittsburgh, Pa.
Wrought iron pipe.

CARRIER CORPORATION, 285 Madison Ave., New York, N.Y.
Refrigeration stores equipment; galley equipment; ships service air-conditioning equipment; vacuum pumps.

CENTURY LIGHTING INC., 521 W. 43rd St., New York 36, N.Y.
Dimmers for main lounge.

CHAPMAN VALVE CO., Broad and Arch Sts., Philadelphia 7, Pa.
Main injector and overboard discharge valves.

CHEM. PUMP CORP., 1313 E. Mermaid Lane, Philadelphia, Pa.
Waste transfer pump.

CHICAGO PUMP CO., 2300 Wolfram St., Chicago, Ill.
Motor-driven sewage pumps.

CLARK-COOPER CO., 516 N. Market St., Palmyra, N.J.
Steam whistle.

CLEAVER BROOKS, 225 N. Grand Ave., Waukesha, Wisconsin.
Evaporating plants.

COLORADO FUEL AND IRON CO., WICKWIRE SPENCER DIV., Claymont, Del.
Steel plates; flanged and disc heads.

COMMERCIAL ENGINEERING, 1627 K St., N.W. Washington 6, D.C.
Purifiers, diesel oil.

COMMERCIAL RADIO SOUND CORP., 652 First Ave., New York, N.Y.
Broadcast antenna systems.

CONTINENTAL ELECTRIC CO., 325 Ferry St., Newark 5, New Jersey.
Motor generator sets.

CRANE CO., 245 Master St., Philadelphia, Pa.
Plumbing fixtures.

CROUSE HINDS CO., 1347 Wolf St., Syracuse, N.Y.
Electric fittings and fixtures.

CUTLER HAMMER, INC., 1391 W. St. Paul Ave., Milwaukee, Wisconsin.
Fire door closing equipment.

DAVIS ENGINEERING, 1064 E. Grand St., Elizabeth, N.J.
Fresh water sample coolers; water heaters; blowdown evaporator.

DETROIT CONTROLS CORP., 5900 Trumbull Ave., Detroit, Mich.
Pressure switches.

L. C. DOANE CO., Essex, Conn.
Electric fixtures.

EASTERN COLD STORAGE INSULATION, 101 Park Ave., New York 17.
Insulation.

ENTERPRISE GALVANIZING CO., Cumberland and Almond Sts., Philadelphia.
Galvanizing.

EXIDE INDUSTRIAL DIV., 101 N. 33rd St., Philadelphia 4, Pa.
Rectifiers.

FAIRBANKS MORSE, 401 N. Broad St., Philadelphia, Pa.
Emergency diesel generator.

FEDERATED METALS DIV., Perth Amboy, N.J.
Lead shielding.

GALBRAITH-PILOT MARINE DIV. OF MARINE ELECTRIC CORP., 600 Fourth Street, Brooklyn, New York.
Salinity equipment; public address equipment; music entertainment systems.

GRISCOLM-RUSSELL CO., Land Title Bldg., Philadelphia, Pa.
Blowdown evaporators.

GRUENDLER CRUSHER AND PULVERIZING CO., 2915 N. Market St., St. Louis, Mo.
Galley equipment.

HENSCHEL CORP., Amesbury, Mass.
Shaft revolution indicator and rudder angle indicator equipment; engine order telegraph equipment; audible signal bells and contacts.

HOSE McCANN TELEPHONE CO., 25th and 3rd Avenues, Brooklyn, N.Y.
Sound powered telephone equipment.

HOTPOINT COMPANY, 5600 W. Taylor St., Chicago 44, Ill.
Galley equipment; electric ranges.

INSINGER MACHINE CO., 6245 State Road, Philadelphia, Pa.
Dishwashing machines.

INTRUSION PREPAKT CO., Western Savings Fund Bldg., Philadelphia, Pa.
Reinforced concrete for secondary shielding.

JONES AND LAUGHLIN STEEL CO., 462 Gateway Center, Pittsburgh, Pa.
Jr. steel shapes.

KEARFOTT CO., 117 Liberty St., New York City, N.Y.
Windows; fixed lights.

KENNEDY VALVE AND MFG. CO., Elmira, New York.
Miscellaneous valves.

WALTER KIDDE CO., Suburban Sta. Bldg., Philadelphia, Pa.
Mn. alarm and fire detection system.

KOVEN FABRICATORS, 98 E. Dickerson St., Dover, N.J.
Fresh water heaters; air tanks.

LAKE SHORE CORP., 17 Battery Place, New York 4, New York.
Cargo and topping and accommodation ladder winches.

LAMSON, INC., 303 Lawrence St., Syracuse 1, New York.
Pneumatic tube equipment.

LOVELL DRESSEL CO., 478 Forest St., Arlington, N.J.
Electric fixtures.

LUKENWELD DIVISION OF LUKENS STEEL CO., 572 Highway Bldg., Coatesville, Pa.
Swimming pool.

LUKENS STEEL CO., 572 Highway Bldg., Coatesville, Pa.
Steel plate.

LUMINATOR, INC., 120 N. Peoria St., Chicago, Ill.
Electric fixtures.

MACGREGOR COMMARAIN, INC., Box 231, Middletown, Pa.
Cargo hatches.

MARINE ELECTRONICS, INC., 1510 Guilford Ave., Baltimore, Md.
Electrical test panel; general alarm panel.

MARLO COIL CO., 6135 Manchester Ave., St. Louis, Mo.
Heating coils.

MAR-OIL ELECTRIC, INC., 219 Hudson St., Hoboken, N.J.
Electric storage battery.

MACKAY RADIO AND TELEGRAPH CO., 15 Lombard St., Philadelphia, Pa.
Radio equipment; radio direction finding equipment; Loran system; lifeboat radios.

MIDVALE HEPPENSTALL CO., Nicetown, Philadelphia, Pa.
Steel forgings.

MINK-DAYTON CO., 117 N. Fendlay St., Bayton, Ohio.
Lighting fixtures.

MINNEAPOLIS-HONEYWELL REG. CO., 3345 W. Hunting Part Ave., Philadelphia 32, Pa.
Auto. heating and air condition control system.

MOBILE PULLEY AND MACHINE WORKS, 1301 Tennessee Ave., Mobile.
Steel castings.

JULIUS MOCK AND SONS, INC., 777 Rutland St., Brooklyn 3, N.Y.
Hatches; manholes; doors.

PAUL MORROW CO., Wilmington, Del.
Pilot house console.

MURLIN MFG. CO., 200 Bk. S. Ambler, Quakertown, Pa.
Electrical fixtures.

NASH ENGINEERING, 254 S. 15th St., Philadelphia, Pa.
Pumps, priming-motor driven.

NATALE MACHINE AND TOOL CO., Broad St. and Highway 17, Carlstadt, N.J.
Floodlights for containment vessel.

NEW ENGLAND TRAWLER EQUIP. CO., 299 Eastern Ave., Chelsea, Mass.
Towline cable reels.

OHIO INJECTOR CORP., Suburban Station Bldg., Philadelphia, Pa.
Valves.

OTIS ELEVATOR CO., 502 N. 19th St., Philadelphia, Pa.
Elevators, stores, cargo and passengers.

PERKINS MARINE LAMP AND HARDWARE, 47 Pitkin Ave., Brooklyn
Visual whistle signal equipment.

PHELPS DODGE COPPER PROD. CO., Yonkers, New York.
Marine cable.

PHOENIX IRON AND STEEL CO., Harrisburg, Pa.
Steel plate.

PROGRESS ELECTRONICS CO., Dept. P-6, 296 Broadway, New York, N.Y.
Adapters, plugs and jacks.

RADIO CORP. OF AMERICA, Front and Cooper Sts., Camden, N.J.
Radio telephone equipment; radar equipment; television equipment.

RANDOLPH POTTS CO., 6745 Torresdale Ave., Philadelphia, Pa.
Grinder (Standard Tool Co.); drill press (Royersford).

J. T. REDDY CO., 162 S. Easton Road, Philadelphia, Pa.
Engine lathe (South Bend).

REMLER CO., 2101 Bryant St., San Francisco, Calif.
F.m. antenna system; dumbwaiter communication system.

ROSS HEAT EXCHANGER DIV. OF AMERICAN STANDARD, Suburban Square Building, Ardmore, Pa.
Supply gland leak-off vapour condensers.

RUDMAN AND SCOFIELD, 277 Pearl St., New York 38, N.Y.
Cooking and commissary equipment.

SANDUSKY FOUNDRY AND MACHINE CO., Sandusky, Ohio.
Centrifugal castings.

SHENANGO FURNACE CO., Dover, Ohio.
Centrifugal castings.

SIMES CO., 114-15 15th St., College Point, Long Island, N.Y.
Illuminated signs and fixtures.

SPERRY PIEDMONT CO. DIV. OF SPERRY RAND CORP., Willoughby St., Brooklyn, New York.
Gyro compass equipment; gyro pilot.
SQUARE D CORP., 2310 Haverford Road, Ardmore, Pa.
Fused disconnect switches.
STERLING BRONZE CO., 211 63rd St., Brooklyn, New York.
Electric fixtures.
SUPERIOR LINDERWOOD MUNDY CORP., 7 Day St., New York 7, N.Y.
Capstans.
TRANE COMPANY, Suburban Station Bldg., Philadelphia, Pa.
Convectors.
U.S. STEEL CORP., Box 2069, Pittsburgh, Pa.
Steel plates, shapes, sheet and bar.
VIKING PUMP CO., 2336 Fairmount Ave., Philadelphia 30, Pa.
Pumps, diesel oil, transfer and booster.
WARREN PUMP CO., 200 Bridges Ave., Warren, Mass.
Pumps, salt water; intermediate heat exchanger.
WASHINGTON ALUMINIUM CO., Penna. Railroad and Knecht Ave., Baltimore, Md.
Accom. ladders.
WELIN BOAT AND DAVIT CO., 500 Market St., Perth Amboy, N.J.
Lifeboats.
WESTERN GEAR CORP., 417 9th Ave., South Seattle, Wash.
Cargo winches.
WESTINGHOUSE ELECTRIC CO., 3001 Walnut St., Philadelphia, Pa.
Main generator, distribution and emergency switchboards; steering power transfer switchboard; distribution panels; polyethylene sheets f/shielding.
C. H. WHEELER, 19th and Lehigh Ave., Philadelphia, Pa.
Windlass.
WILLIAMS, BROWN AND EARLE, 904 Chestnut St., Philadelphia, Pa.
Sound motion picture equipment.
L. J. WING MFG. CO., 55-D Vreeland Mills Road, Linden, N.J.
Vent fans.
ALAN WOOD STEEL CO., Wood Road, Conshohocken, Pa.
Steel plate.
WORTHINGTON PUMP CO., 426 Worthington Ave., Harrison, N.J.
Pumps, submersible bilge, motor driven.
YOUNGSTOWN SHEET AND TUBE CO., 1954 Stambaugh Bldg., Youngstown, Ohio.
Steel plates.

MAJOR VENDORS TO BABCOCK AND WILCOX CO.

A. MAJOR REACTOR SYSTEM COMPONENTS

AIR REDUCTION CO., Charlotte, N.C.
Hydrogen addition system.
ALLIS-CHALMERS MFG. CO., Milwaukee, Wis.
Primary coolant pumps.
BUFFALO FORGE, Buffalo, N.Y.
Containment vessel air-conditioning system.
CHEMPUMP CO., Philadelphia, Pa.
Emergency cooling pump.
COCHRANE CO., Philadelphia, Pa.
Primary demineralizers.
CRANE CO., Chicago, Ill.
(a) Primary gate valves.
CUNO ENGINEERING, Meridan, Conn.
Effluent filters.
EDWARDS VALVE CO., E. Chicago, Ill.
Main steam valves; (*b*) Primary check valves.
GARY STEEL CO., Lynchburg, Va.
Buffer surge tank.
GENERAL ELECTRIC, Nuclear Products Division, San Jose, Calif.
Control rod drive system.
GRAHAM MFG. CO., Batavia, N.Y.
Letdown coolers; emergency cooler.
JOHNS-MANVILLE CO., Philadelphia, Pa.
Insulation.
KNAPP MILLS CO., Wilmington, Del.
Primary shield tank.
ARTHUR D. LITTLE CO., Cambridge, Mass.
Gaseous waste collection system.
MALLINCKRODT CHEMICAL, St. Louis, Mo.
Fuel in powder form.
MANNING, MAXWELL AND MOORE, Stratford, Conn.
Pressurizer relief valves; boiler safety valves.
Constructed by NEW YORK SHIPBUILDING CORP., according to Babcock and Wilcox design parameters.
Containment vessel and shielding.
RICHMOND ENGINEERING, Richmond, Va.
Intermediate system surge tank; effluent condensing tank.
STEIN SEAL CO., Philadelphia, Pa.
Buffer seals.
STRUTHERS-WELLS, Warren, Pa.
Intermediate coolers; buffer seal coolers.
E. L. WIEGAND, Pittsburg, Pa.
Pressurizer heaters.
WORTHINGTON CORP., Harrison, N.J.
Buffer charge pumps; booster pumps.

B. STEAM PROPULSION SYSTEM COMPONENTS

V. D. ANDERSON, Cleveland, Ohio.
*Moisture separator.
DAVIS CO., Elizabeth, N.J.
*Low pressure feedwater heater.
DELAVAL STEAM TURBINE CO., Trenton, N.J.
Steam turbines; main reduction gear; turbine generators; turbines; lube oil service pumps (IMO).
DELAVAL SEPARATOR CO., Poughkeepsie, N.Y.
*Lube oil purifier.
G.E., Schenectady, N.Y.
*Turbine generators; generators.
GIMPEL MACHINE WORKS, Philadelphia, Pa.
*Manoeuvring valve.
INGERSOLL RAND CORP., Phillipsburg, N.J.
*Main and auxiliary condensers; main and auxiliary air ejectors.
WESTINGHOUSE, Pittsburgh, Pa.
Electrical switchgear.
WORTHINGTON CORP., Harrison, N.J.
*Main circulating pumps; main turbine-driven feed pumps; main and auxiliary condensate pumps; de-aerating feedwater heater.
* By subcontract from DeLaval Steam Turbine Company.

C. AUXILIARY ENGINE ROOM MACHINERY COMPONENTS

CLEVELAND DIESEL DIVISION, GENERAL MOTORS CORPORATION
Diesel generator sets.
CYCLOTHERM CO., Oswego, N.Y.
Emergency package.
DRAVO CO., Ellicot City, Md.
Low pressure steam generator.
GENERAL ELECTRIC CO., Schenectady, N.Y. (Subcontract to DeLaval.)
Take-home motor.

D. INSTRUMENTATION AND CONTROL PANEL COMPONENTS

BAILEY METER CO., Cleveland, Ohio.
Non-nuclear instrumentation and controls; main control panel and two auxiliary control panels.
WESTINGHOUSE, E. Pittsburgh, Pa. (Subcontract to Bailey Meter Co.)
Nuclear instrumentation; radiation monitoring.

E. MAINTENANCE AND HANDLING EQUIPMENT COMPONENTS

DIAMOND POWER SPECIALITY CORP., Lancaster, Ohio.
Stud tensioner.

YANKEE

Main Contractor
Westinghouse
Electric

ALLIS-CHALMERS MANUFACTURING COMPANY., 1160 Park Square Building, 31 St. James Ave., Boston 16, Mass.
Purification, cooling and drain pumps.
AMERICAN FILTER CO., INC., J. M. Pettingell Associates, 385 Concord Ave., Belmont 78, Mass.
Incinerator gas scrubber.
AMERICAN INSTRUMENT CO., Amino Sales Corp. of Mass., Agent, 9 Clark Road, Bedford, Mass.
Waste gas compressors.
AMES IRON WORKS, INC., N.L.C. Engineering and Equipment Co., Agents, 101 Tremont St., Boston 8, Mass.
Auxiliary steam boilers.
ARTISAN METAL PRODUCTS, INC., 73 Pond St., Waltham 54, Mass.
Incinerator.
THE BABCOCK AND WILCOX COMPANY, Barberton, Ohio.
Reactor vessel; pressurizer.
BINGHAM PUMP COMPANY, 201 Devonshire St., Boston 10, Mass.
Low pressure surge tank cooling and drain cooling pump; shutdown cooling pump; safety injection pumps.
BYRON-JACKSON PUMPS, INC., 980 Worcester St., Wellesley Hills 81, Mass.
Service water pumps.
CHICAGO BRIDGE AND IRON COMPANY, 201 Devonshire St., Boston 10, Mass.
Vapour container.
ENGINEERS AND FABRICATORS, INC., 3501 W. 11th St., Houston, Texas.
Shutdown cooling heat exchanger; fuel pit cooling heat exchanger.
FAIRBANKS, MORSE AND CO., 178 Atlantic Ave., Boston 10, Mass.
Gravity tank transfer pumps.
HUNGERFORD AND TERRY, INC., Clayton, N.J.
Water treatment plant.
INGERSOLL-RAND COMPANY, 285 Columbus Ave., Boston 16, Mass.
Boric acid solution transfer pumps; boric acid mixing and storage tank; monitored tank transfer pumps; waste liquid transfer pumps; stripper bottoms pumps; distillate pump; test tank effluent pump.
KIMBALL-JONES (PEERLESS PUMP DIVISION, FOOD MACHINERY AND CHEMICAL CORP.), 150 Causeway St., Boston 14, Mass.
Fire pumps.
MISSOURI BOILER AND TANK COMPANY, Division of Nooter Corporation, 23rd and Papin Sts., St. Louis 3, Mo.
Waste holdup tank; activity dilution decay tank; primary water storage tank.
MANNING, MAXWELL AND MOORE, INC., Bridgeport 2, Conn.
Main coolant safety valves; travelling cranes.
NATIONAL VALVE AND MANUFACTURING COMPANY, 3101 Liberty Ave., Pittsburgh 1, Pa.
Main coolant piping.
PANTEX MANUFACTURING CORPORATION, 521 Roosevelt Ave., Pawtucket, Rhode Island.
Evaporator.
PORTLAND COMPANY, Div. of Airtex Products, Inc., 58 Fore St., Portland 6, Maine.
Reactor shield tanks.
JAMES RUSSELL ENGINEERING WORKS, INC., Dewar and Auckland Sts., Boston 25, Mass.
Purification ion exchangers; fuel pit ion exchanger.
SOUTHWESTERN ENGINEERING COMPANY, 4800 Santa Fe Ave., Los Angeles 58, Calif.
Stripper overhead condenser; evaporator overhead condenser; compressor suction cooler; compressor discharge cooler; feed and bleed heat exchanger.
STRUTHERS WELLS CORPORATION, c/o Mr. T. F. Fitzgerald, Representative, 201 Devonshire St., Boston 10, Mass.
Component cooling heat exchangers.
TOWER IRON WORKS, 50 Borden St., P.O. Box 1314, Providence, Rhode Island.
Waste disposal tanks.
UNITED SHOE MACHINERY CORPORATION, Atomic Power Department, Beverly, Mass.
Fuel handling system.
UNION PUMP SALES COMPANY, Neal W. Cargill, Representative, 185 Devonshire St., Boston 10, Mass.
Charging pumps.
WORTHINGTON CORPORATION, 150 Causeway St., Boston 14, Mass.
Component cooling pumps.

TRANSLATIONS

A Résumé of Principal Articles

L'Energie Nucléaire aux Etats-Unis

Les efforts déployés par les Etats-Unis pour le développement de l'exploitation pacifique de l'énergie nucléaire sont d'un ordre de grandeur bien supérieur à ceux de n'importe quel autre pays occidental et probablement du monde entier. L'avance n'a pas seulement lieu sur un large front, mais elle se manifeste également en profondeur. Pour la passer en revue, il faut effectuer un choix et se résoudre à passer sous silence certaines réalisations. Le désir d'être à la pointe du progrès dans le maximum de directions possibles s'est manifesté, par example, par la décision de mettre en chantier le " Savannah " qui figure au centre de notre exposé. Il ne s'agit même pas d'un prototype de bateau économique. Mais sa construction et son utilisation contribueront dans une forte mesure à accroître nos connaissances des nombreux problèmes posés par la propulsion maritime nucléaire.

Les réacteurs refroidis et modérés à l'eau ont été à la tête des réalisations et recherches américaines dans le domaine des réacteurs. Les centrales civiles les plus importantes de ce type sont Dresden et Yankee. Elles sont équipées, la première d'un réacteur à eau bouillante, la seconde d'un réacteur à eau sous pression. Elles vont se metre d'ici peu a alimenter en courant le secteur public. Bien que les réacteurs à eau bouillante soient dé à bizn connus dans leurs principes, l'expérience qui sera aiguise par l'exploitation des centrales de Dresden et de Yankee aura certainement une influence notable sur leur futur development. D'autres systèmes sont également étudiés sérieusement dont certains ne seront probablement intéressants qu'à long terme. Les réacteurs à modérateur organique ont de sérieuses prétentions dans un avenir proche. Les réacteurs à eau lourde autrefroidis par un gaz ainsi que les réacteurs rapides, entre autres, semblent devoir être interessants à plus longue échéance. Il a été nécessaire, à cet égard, d'ajuster le programme américain pour le rendre plus rationnel.

Mais en plus de la construction de centroles de base de forte capacité, les Etats-Unis ont en outre lancé un programme relatif à l'utilisation de l'énergie nucléaire pour le fourniture d'électricité à des localités ou installations isolées et d'accessibilité difficile ou à des véhicules spatiaux. Dans ce domain ne ils font pratique cavalier seul. Des recherches très poussées sont en cours: les équipements requis doivent, en particulier, être très compacts et démontables.

De vastes installations sont nécessaires pour l'étude de l'énergie thermonucléaire. Mais, tenant compte de la nouveauté des problèmes quise pusent en physique des hautes températures, le mode d'attaque actuel concentre les efforts sur l'étude des principes fondamentaux plutôt que sur des expériences.

Plusieurs pays recevront des renseignements ou de l'équipment des Etats-Unis. La politique intérieure et extérieure de la commission à l'énergie atomique et de l'industrie nucléaire américaines auront toutes deux de l'influence sur les études et les réalisations effectuées dans le reste du monde. L'industrie américaine assurée de débouchés futurs pour ses produits, en Europe, au Japon, en Amérique et ailleurs a une periode difficile devont elle. Sa puissance future dépendra dans une grande mesure, des solutions plus ou moins convenables qu'elle aura données aux problèmes qui lui sont actuellement posés.

Die Kernkraft in den Vereinigten Staaten von Amerika

Die Arbeit, die die Vereinigten Staaten der Ausnutzung der Anwerdung der Atomkraft zu Zwecken des Friedens gewidmet haben und die bei weitem viel grösser ist, als die irgend eines anderen Staates in der Welt des Westens (und wahrscheinlich sogar in der ganzen Welt), ist nicht nur begrifflich sehr weit gefasst, sondern geht auch sehr in die Tief der Probleme. Bei jeder Uebersicht muss man daher äusserst wählerisch vorgehen und manche Seiten der Kernkraftentwicklung übergehen. Der Wunsch in möglichst vielen Zweigen als führend anerkannt zu werden, ist die hauptsächliche Triebkraft in Amerika, und dies ist ein typischer Grund für die Entscheidung, die " Savannah " zu bauen, die im Mittelpunkt der Uebersicht steht. Das Schiff ist nicht gerade nur als ein Vobild für ein wirtschaftliches Seefahrzeug gedacht, sondern seine Konstruktion und sein Betrieb werden in grossen Masse den Weg zu einem internationalem Verständnis der zahlreichen Probleme bahnen, die bei dem Antrieb durch die Kernkraft auftreten.

Die mit Wasserkühlung und -Moderierung arbeitenden Systeme stehen an der Spitze der Entwicklung der amerikanischen Reaktoren, von denen die ersten nichtmilitärischen Anlagen grossen Stiles Dresden und Yankee sind. Diese beiden Reaktoren, der erstere ein Siedewasser-Reaktor, der andere ein Druckwasser-Reaktor, werden demnächst elektrischen Strom in die Netze der Nachbarschaft liefern, und obwohl bei Siedewasser-Reaktoren bereits die Grundsätze der Systeme sich als richtig erwiesen haben, so werden doch die Betriebserfahrungen bei Dresden und Yankee einen grossen Einfluss auf die weitere Entwicklung ausüben. Andere Systeme werden ebenfalls energisch verfolgt mit Einschluss der sogenannten Projekte auf lange Sicht. Das Programm ist in gewissen Masse rationalisiert worden, jedoch steht die Moderierung mit organischen Stoffen in starkem Wettbewerb für die sofortige Ausführung, während der gasgekühlte, mit schwerem Wasser arbeitende schnelle Reaktor ebenso wie noch andere Typen reizvolle Möglichkeiten bieten.

Aber abgesehen von grossen Werken mit hoher Grundbelastung. sind die Vereinigten Staaten fast allein mit ihrem Programm, die Kernkraft zur Antriebsleistung für weit entfernte Orte und für Weltraum-Fahrzeuge heranzuziehen mit dem Ergebnis, dass ausgedehnte Untersuchungen in Bezug auf die technischen Probleme gemacht werden, die sich aus dem Zwang zur Kompaktheit und zur Demontierbarkeit ergeben.

In Hinsicht auf die thermischen Probleme des Atomkerns ist das Arbeitsgebiet enorm, jedoch konzentriert sich im Augenblick die Arbeit in Erwägung der absoluten Neuheit der Physik der hohen Temperaturen vielmehr auf das Studium der Grundlagen als auf Experimente, die sich mit den vorliegenden tatsächlichen Dingen befassen.

Viele Länder werden von dem Programm der Vereinigten Staaten inbezug auf Information und Lieferung von ausgeführten Teilen abhängig sein, und sowohl die innere, als auch die auswärtige Politik der Atomic Energy Commission und der Industrie der Vereinigten Staaten werden die Entwicklung und die Ausnutzung in Ueberseegebieten beeinflussen. Die amerikanische Industrie wird, obwohl sie mit Vertrauen in die Zukunft der Märkte für ihre Produkte in Europe, Japan, im Inneren Amerika und ausserhalb sieht, eine schwierige Periode vor sich haben, und ihre Stärke wird von einer erfolgreichen Lösung der Probleme abhängen, die sie vor sich sieht.

La Energía Nuclear en los Estados Unidos.

Mucho más importantes que los de cualquier otro pais del Mundo Occidental (y quizás del mundo entero), los esfuerzos de los E.E.U.U. dedicados a la explotación de energía atómica para aplicaciones pacíficas, son no solo amplios en concepto sino también de considerable profundidad. Por lo tanto, cualquier tentativa de pasarlos en revista tiene que ser altamente selectiva, dejando sin mencionar muchos aspectos de la evolución nuclear. En los E.E.U.U. uno de los mayores impulsos ha sido el deseo de que se reconozca su posición de vanguardia en tantos ramos como sea posible, y este aspecto está representado por la decisión de construir el " Savannah " de que trata nuestro artículo principal. Este buque no es ni siquiera un prototipo de buque económico, pero su construcción y su operación mucho harán en el sentido de preparar el camino para un entendimiento internacional con respecto a los muchos problemas que hay que enfrentar en abordar la propulsión nuclear.

Formando la "punta de lanza" del desenvolvimiento americano de reactores han estado los sistemas enfriados por agua y moderados, siendo las primeras plantas civiles de gran envergadura el Dresden y el Yankee. Estos dos reactores, el primero un BWR y el segundo un PWR, pronto estarán alimentando electricidad dentro de sistemas de redes circunvecinos y aunque los reactores de agua hirviente ya han comprobado los principios de los sistemas, la experiencia en operación de Dresden y Yankee ejercerán un gran influjo sobre desenvolvimientos futuros. También se están examinando enérgicamente otros sistemas, incluyendo los que se sabe van a ser proyectos a largo plazo. Ha ocurrido alguna racionalización del programa pero la moderación orgánica es una fuerte contendora para honores inmediatos, y los reactores enfriados por gas, agua pesada y rápidos así como también otros diversos tipos poseen atractivas potencialidades.

Pero aparte de las centrales de carga básica grande, se puede decir que los Estados Unidos es casi la única nación que se ha lanzado a la implementación de un programa para el aprovechamiento de energía nuclear en el suministro de fuerza a lugares remotos y a vehículos astronáuticos, con el resultado que se están llevando a cabo extensas investigaciones con respecto a las implicaciones ingenieriles de compactabilidad y desmontabilidad.

Por el lado termonuclear, la escala de operaciones es vasta; pero, con acertado aprecio de lo que vale la completa novedad de la física de elevadas temperaturas, las actuales investigaciones han sido concentradas en estudios de los principios fundamentales más que en experimentos sobre una base " ad hoc."

Muchos países van a depender del programa de los E.E.U.U. para la provisión de datos y de equipo, y las políticas, tanto internas como de Ultramar, de la AEC y la industria de los E.E.U.U. afectarán los desenvolvimientos y la explotación de los países de Ultramar. La industria de los E.E.U.U., mientras que tiene confianza en los futuros mercados para sus productos en Europa, Japón, dentro de los mismos E.E.U.U. y en otras partes, está enfrentando un periodo difícil y su fuerza depende de una solución venturosa de los problemas con que tiene que lidiar.

Industrial Notes

Britain's second nuclear submarine is to be built at Barrow by **Vickers-Armstrongs** with propulsion machinery supplied by **Rolls-Royce and Associates.** Although similar to *Dreadnought*, which is due to be launched by H.M. the Queen on October 21, the second ship will be larger in size and its propulsion unit will be based on the prototype, at present under construction at Dounreay and due to begin operation in 1962. On the basis of present plans the primary circuit will be in mild steel.

The former subsidiary of G. A. Harvey, **Nuclear Engineering, Ltd.,** has been acquired by **Hawker Siddeley Nuclear Power,** although the company's previous activities in the fields of medical therapy and industrial radiography are to be taken over by **Test Equipment (Models)** and **Panatron** respectively. This will leave Nuclear Engineering, Ltd., free to concentrate upon the design and production of irradiation equipment for research organizations and industry where ultimate markets for sterilization and processing equipment are expected to be high.

Reactor control ionisation chambers for incorporation in General Atomic's Triga reactors are being supplied by **20th Century Electronics.** Amongst a large order recently delivered are the type RC-6 gamma-compensated chambers based on original Harwell designs.

CNRN in Italy has ordered analogue computing equipment from **Miles Electronics** for Rome University, where it will be used primarily for training purposes. A similar installation has been ordered by the Kjeller research establishment in Norway: it will be used in conjunction with a general purpose computer.

The complete nuclear instrumentation for the AEA's AGR at Windscale is to be supplied by **Plessey Nucleonics.** The company had previously received a contract for the supply of the BSD equipment for this same reactor.

A new building housing the main library and document reproduction section has been opened at **Harwell.** The total floor area of the new facility is 40,000 sq. ft and the building cost is estimated at £200,000.

AEI is to supply a 4-MV liner accelerator and equipment for housing a 2,000 c cobalt source to each of two hospitals in Glasgow. At Belvidere where work is further advanced than at the Western Infirmary the first source irradiation equipment is already being installed in temporary quarters.

Following the agreement last year between **C. and J. Weir** and Clark Brothers of Dresser A.G. which provided for the manufacture in Scotland of Clark centrifugal and reciprocating compressors, arrangements have now been made for the similar manufacture of Isotemp packaged centrifugal compressors with capacities of 5,000 to more than 50,000 ft³/min.

A new British Standard, B.S. 3258:1960 has been prepared to cover silicone rubber insulated cables and flexible cords.

The name of **British Refrasil** of Sillington, Co. Durham, a subsidiary of **Darlington Chemicals,** is to be changed to **Darchem Engineering.** Ownership and the functions of the company remain unaltered, and the division supplying Refrasil insulation will carry the name British Refrasil Division.

A 400-ton capacity Goliath in course of erection at APC's site, Trawsfynydd.

Dewrance announce that their subsidiary company **Jones Tate** of Bradford is being transferred to Gt. Dover Street, London.

The new works and offices of **Henry Hargreaves** at Lord Street, Bury, which have now been officially opened include two additional manufacturing bays of 10,000 sq. ft. The company has supplied a variety of equipment to the AEA's establishments.

Matthews and Yates, fan engineers and **Taylor and Brown** chemical plant engineers are to manufacture jointly under the trade name **Turbro-Cyclone,** a wide range of plastic chemical plant and equipment.

The 10th Electrical Engineers Exhibition will open at Earls Court on March 21, 1961, for five days.

The recovered capsule from the Discoverer satellite which made 17 revolutions round the earth contained nuclear emulsions supplied by **Ilford.** The company recently produced a stack of sheet emulsions weighing 600 lb.

Personals

Appointments

Mr. R. L. J. Hayden, Mr. C. J. Geiss, Mr. J. F. G. Arman, Mr. A. S. Bridgwater and **Mr. J. McLean** to the board of Foster Wheeler.

Mr. F. S. Barton to the board of Mullard Equipment.

Lord Citrine, Sir James Chadwick and **Mr. C. F. Kearton** have been reappointed as part-time members of the U.K. AEA.

Mr. H. R. Schofield as chief transmission project engineer to the CEGB. **Mr. P. P. Fowler** as civil engineer (transmission).

Mr. G. C. Roberts as commercial director of Fairey Engineering.

Mr. W. G. Harrison as general manager, Dynatron Radio.

Mr. W. A. Ankerson as assistant managing director of Associated Electrical Industries (Woolwich).

Mr. J. B. Adams, director, CTRE, Culham, as director general of CERN until July 31, 1961.

Commander E. G. Sutton as works director of Sir W. H. Bailey and Co.

Mr. R. Turner as managing director of Powell Duffryn Technical Services.

Mr. J. R. Erskine as sales director of Aveley Electric.

Mr. Edwin Baden as chairman, **Mr. R. R. Baden** as deputy chairman, **Mr. P. D. Doulton** and **Mr. S. Kindler** as managing directors, **Mr. R. M. Speir** as director of Matthew Hall.

Sir Donald Perrott, K.B.E., as director and deputy chairman of Dewrance.

Sir Joseph Latham, C.B.E., as director of A.E.I. **Mr. C. R. Wheeler,** C.B.E., as additional vice-chairman of A.E.I.

Mr. C. H. T. Williams and **Mr. E. G. Plucknett** to the board of Tube Investments.

Overseas

John R. Moor as assistant manager for administration, U.S. AEC's Oak Ridge Operations.

G. L. Fox and **D. R. Roberts, Jr.,** to Allis-Chalmers Manufacturing Company's nuclear power department.

Milton F. Pravda as chief of system design with the Martin Company's Nuclear Division.

Mr. C. E. H. Verity as director of Foster Wheeler Corporation of New York.

J. E. Mahlmeister as leader of SRE group, Atomics International.

Mr. Vyacheslav M. Molotov replaces **Mr. Leonid M. Zamyatin** as representative of the U.S.S.R. to IAEA.

Howard S. Bunn as vice-chairman and **Birny Mason, Jr.,** as president of Union Carbide Corporation.

Dr. Merlin D. Peterson as deputy associate laboratory director at Argonne National Laboratory.

William H. Heath as director of Systems Division, General Precision Inc.

Resignations

Rt. Hon. the Viscount Head of Throope as director of A.E.I.

Mr. J. B. Hamilton-Meikle from the board of Henry Balfour.

Mr. D. G. Hemmant of Powell Duffryn Technical Services.

Retirements

Mr. L. B. Robinson and **Mr. R. G. H. Linzee** from the board of British Aluminium Company.

Mr. G. A. S. Nairn as part-time member of CEGB.

Obituary

Nuclear Engineering regret to report the death of: Lord Weeks, chairman of Vickers, Ltd., from 1948 to 1956. After resigning from this position he retained his seat on the board, with special responsibility for nuclear development.

Processes and Equipment

For further information on any item please enter the relevant number on the Reader Service Card enclosed with this issue and forward the card to the address given.

The Richardsons Westgarth expansion joint undergoing life tests.

New Duct Expansion Joint

A new type of expansion joint, for gas ducts, developed by Richardsons Westgarth, is to be used for the first time at Trawsfynydd (A.P.C., Ltd.). As will be seen from the diagram, the two halves of the joint are connected together by two sets of high-tensile steel rods, spaced around the periphery of the duct so that each set forms the frustum of a cone. In the axial direction, the joint behaves as a stiff connection, but angular deformation can be accommodated by the transverse bending of the rods. Since movement is entirely due to pure elastic deformation of the rods, the forces involved are both accurately predictable, and do not vary with time. By varying the basic dimensions of the restraint (i.e., the number and size of the rods and the cone angle) a wide range of working deflections and restoring moments can be obtained.

The two ends are connected by a bellows of conventional design, inside which is a centring device, thus ensuring that no torque loading can be applied to the bellows and equalizing the deflections of the two sets of bars under conditions of steeply changing bending moment along the duct. The bellows are also relieved of all shear forces accompanying such a change in bending moment.

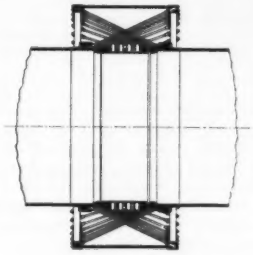


Diagram of arrangement of rods.

Apart from the bellows, the rods are the only portion of the joint which are subject to any appreciable stresses, and, since the stress system is amenable to accurate calculation, an approach to an ideal shape of rod is possible. The rods are cold swaged to a profile which approximates constant stress distribution along their length; selected numbers undergo fatigue tests.

In addition to the obvious advantages of a structure which is equally flexible in any direction, and the elimination of fretting or seizure which could occur with pin-jointed designs, the gymbal characteristics of the joint offer greater freedom to the station designer, particularly from the viewpoint of civil engineering work.

(Richardsons, Westgarth & Co. Ltd., P.O. Box 2, Wallsend, Northumberland.)

1359

Beryllium Survey Monitor

Beryllium can be detected in a sample containing less than 0.01% BeO, by a radiation monitor jointly developed by Plessey Nucleonics, in conjunction with the U.K. AEA. Neutrons released by the beryllium, upon irradiation with a gamma source, are slowed down in an oil tank, and detected by proportional counters, the sensitivity being such that the presence of 0.004% BeO doubles the background count rate.

The equipment, which is powered by eight U2 dry cells, is transistorized, shockproof, and fitted with adjustable carrying handles to ensure against over-exposure of the operators to radiation. It can be comfortably transported by two men.

(Plessey Nucleonics Ltd., Weedon Road, Northampton.)

1360

Automatic Beta Counting

A major advance in the automatic counting of beta-emitting solid samples is announced by the Nuclear-Chicago Corporation, of Illinois. Main features of the equipment are the extremely low background and the "Micromil" window on the gas-flow detector. The window density, being less than 150 $\mu g/cm^2$, allows the entry of soft betas into the counting chamber, without appreciable loss, and the low background makes it possible to measure samples with an activity as low as 2-3 counts/min.

(Continental Distributors, Ltd., 121 Earls Court Road, London, S.W.5.)

1361

Linear Count Ratemeter

Reliability and low drift are two of the good qualities claimed for a new G.E. linear count ratemeter with a range of 0—10^7 counts/min, in ten overlapping ranges. Indication of the average rate of arrival of random pulses is given on an expanded direct-reading linear scale, with an accuracy of ± 2%, and a zero drift of less than 1%. An adjustable trip circuit, variable from the front of the chassis, operates alarm or protective circuits; a discriminator adjustment allows rejection of background noise. The input will accept pulses as low as 50 mV.

(General Electric, Schenectady 5, N.Y.)

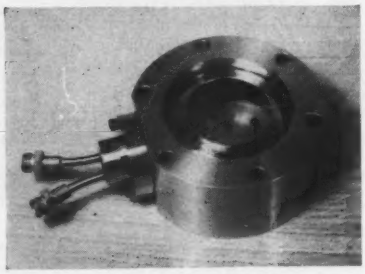
1362

Left: The Plessey beryllium survey monitor.

Right: The G.E. linear ratemeter.

New Vacuum Equipment

Designed for use with the "Speedivac" 2-in. diffusion pump, a new cold baffle, the CB2, has been developed to improve performance and allow of obtaining higher



The new CB2 cold baffle.

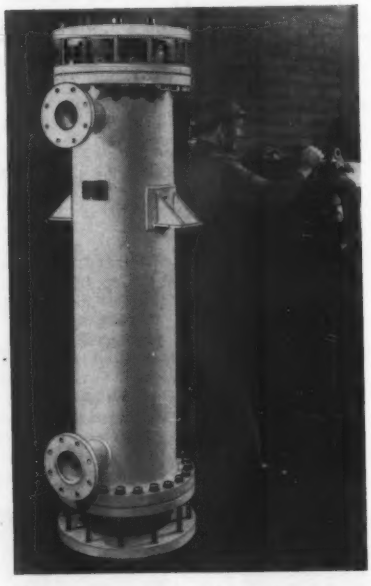
vacua. It is normally supplied with the top flange grooved for an O-ring seal, but for very high vacua and bakeable systems, a plain flange is supplied for the usual aluminium wire gasket.

(Edwards High Vacuum, Ltd., Manor Royal, Crawley, Sussex.)

1363

Graphite Condensers

Problems involving the condensing of corrosive vapours are assisted by a new graphite condenser, latest addition to the Powell Duffryn "Delanium" range. The condensing element consists of a graphite cartridge, with longitudinal holes for the passage of vapour, and an outer surface



Powell Duffryn graphite condenser.

which is slotted to permit a counter flow of cooling fluid between the cartridge and the steel jacket which encloses it. The inlet and outlet heads of the cartridge are of carbon; the jacket may be lined with inert material if cooling fluids other than water are used.

The new unit has considerable advantages in compactness, as compared with the use of glass; it is stated that 1 ft² of graphite is equivalent to 3-4 ft² of glass. It is being manufactured in sizes of 4-90 ft².

(Powell Duffryn Carbon Products, Ltd., Springfield Road, Hayes, Middlesex.)

1364

Small Submerged Filter

The advantages of both self-priming and centrifugal systems are claimed for a new submersible pump and filter unit developed for filtering corrosive or expensive solutions without loss. Lucite and epoxy resin, with stainless steel fittings are standard materials

Left: The Sethco submerged pump.

Right: Savage & Parsons 12-channel oscillograph.

for the pump and filter chamber; for applications over 140°F stainless steel or Teflon may be substituted, while Hastelloy C or titanium may replace stainless steel for high-chloride applications. Two sizes are available, 50-75 and 100-150 gal/h, the same pump being used in both. The open-circuit capacity of the pump is 360 gal/h, with a shut-off pressure of 17 p.s.i.

(Sethco Manufacturing Corporation, 2284 Babylon Turnpike, Merrick, New York.)

1365

Direct-writing Oscillograph

Recording at up to 1,000 c/s, with eight paper speeds from 5 in./min-150 in./sec, a new 12-channel Savage and Parsons oscillograph is actually capable of giving a well-defined trace at writing speeds of above 12,000 in./sec. Recording is on 6-in.-wide sensitized paper, by u/v light from a compact-source lamp, via 15 mirror galvanometers, the additional three being normally used for event or time marking, but can be used for extra traces if desired.

The paper, in rolls up to 200 ft, can be loaded through the front panel, and the trace is spontaneously developed in a few seconds. Paper conservation is aided by a selector mechanism which, if continuous running is not required, cuts off the paper feed after 2, 5 or 10 ft. Speeds, which are push-button selected, may be changed during a run.

Trace adjustment can easily be made by screwdriver, and unwanted traces may be directed off the chart. Identification is

obtained by sequential interruption of each trace at 2-ft intervals, regardless of speed.

Five types of galvo. are available, the standard type having a sensitivity of 7.5 mA/cm, with a tolerance of ±1%, and response up to 1,000 c/s, but galvos. responsive to 2,000 c/s can be supplied.

(Savage and Parsons, Ltd., Otterspool Way, Watford By-pass, Watford, Herts.)

1366

BRIEFLY . . .

Taylor Electrical Instruments, Ltd., Montrose Avenue, Slough, Bucks, announce a new Multimeter, the Model 105a, that combines robust construction with a sensitivity of 20,000 ohm/volt. With a 5-in. scale, calibrated in three colours, to facilitate range identification, the instrument has a total of 21 self-contained ranges; six current ranges (60 μA-12A) seven d.c. voltage ranges (0.3V-1,200V), five a.c. voltage ranges (12V-1,200V), and three resistance ranges (0-20 megohms). A high voltage adapter can be provided for readings up to 30 kV. **B170**

The successful welding of Magnox to stainless steel, by the U.K. AEA, is announced by **British Oxygen Gases, Ltd.,** North Circular Road, Cricklewood, London, N.W.2. The technique was evolved by their Sales Technical Service, and uses "Tinflux" and the "Saffire" lightweight oxy-acetylene blowpipe. **B171**

New electronic control equipment for the precise control of small spot welding machines has been developed by **Lancashire Dynamo Electronic Products, Ltd.,** Rugeley, Staffs. The equipment is designed to provide accurate synchronous timing from ½ to 5½ cycles in half-cycle steps, with a full range of stepless heat control. It is intended for machines up to 15 kVA rating but can be used in connection with Ignitrons, for the largest machines, if necessary. **B172**

Although not yet commercially available, a new synthetic binder has been announced by the **Armour Research Foundation** of Illinois Institute of Technology, 35 West 33rd Street, Chicago, 16, Illinois. The new binder eliminates baking, and it is said that the production time needed to make moulded graphite has been cut from 90 days to less than two hours. **B173**

A new bending machine for bench or pedestal mounting is announced by **Chamberlain Industries, Ltd.,** Staffa Works, Argall Avenue, Leyton, London, E.10. Angles up to 2½ x 2½ by $\frac{5}{16}$ in., flat bars up to 4 x ½ in., gas pipes up to 2 in. bore, or solid bar up to 1¼ in. dia., can be bent cold. **B174**

The makers of "Dust Barrier," **Anti-Dust Services, Ltd.,** of P.O. Box 28, 53a Stafford Street, Dudley, Worcs, announce changes in the formulation of the impregnant. This impregnated net material, which can be hung up on wires to make dust barriers, is now inflammable only in the presence of a flame. **B175**

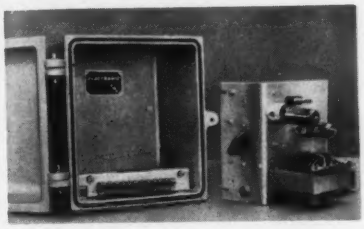
Elcontrol, Ltd., Wilbury Way, Hitchin, Herts. announce the standardization of two alternative forms of automatic weight control units, the LC4 and LC5, the latter having, in addition to an adjustable trip point and an adjustable differential, a trip margin indication. The range of load cells has now been developed from 20 lb. up to 2,000 tons. **B176**

Servomex Controls, Ltd., Crowborough, Sussex, have developed a precision speed control system, type MC 47. The motor, which has a maximum output of 0.5 h.p., can be varied from zero to 7,500 r.p.m. in either direction with a stability of approaching 0.1%. **B177**

The metal finishing division of the **Pyrene Co., Ltd.,** Great West Road, Brentford, Middlesex, announce a new range of treatments for aluminium and its alloys. Additions to the bonderizing range of treatments include Bonderite 710, 701 and 740, and a new controlled alkaline etching process, Aluma Etch 391. **B178**

Capacity-operated Monitor

The Contimitor, a capacity-operated unit which can be applied to level indication, thickness gauging, or difficult counting problems, has been developed to combine robust construction with sensitivity. A crystal oscillator circuit, compensated for temperature and mains voltage variation, employs a



Left: The Contimitor removed from its case.

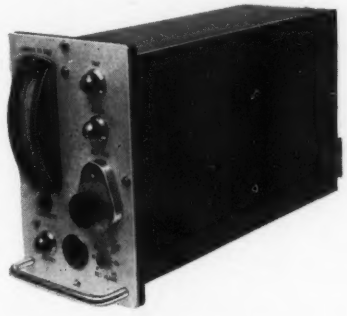
matched transmission line to connect to the probe or transducer, and connections up to 100 yards can be used without loss of stability, or errors from adjacent metal. Normal sensitivity is 1 mA change in output for a capacity change of 10pF; a d.c. amplifier can be applied if necessary, or the output fed to an indicator or recorder. The housing is of aluminium alloy, and the unit is quickly removable without disturbing circuit wiring.

(Electronic Machine Co., Ltd., Mayday Road, Thornton Heath, Surrey.)

1367

Transistorized Trip Amplifier

The Bailey type WB3T trip amplifier Series II, is a rather more sophisticated version of a design which has been widely used at Calder, Chapelcross, Dounreay, Winfrith and Harwell, and is being used at Berkeley. In addition to its function of tripping at a given temperature, it gives visual indication of temperature over a preset range below the set-point, and a warning alarm which can be set to operate anywhere inside this range.



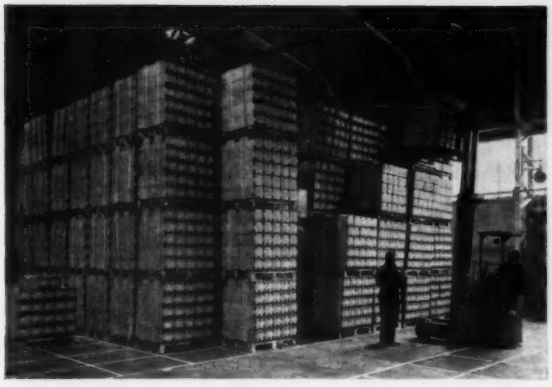
The new Bailey trip amplifier.

Printed circuits are used throughout, while magnetic amplifiers and transistors are used for maximum reliability. Operated from either a.c. or d.c., the instrument has very good tolerance for supply variation, a voltage change from 21-28 (normal 24V) giving only 24 μV change in trip point. Setting of the trip point (i.e., balancing the thermocouple input against a reference voltage) is by a helical potentiometer, the resolution being better than 0.1%. Long-term stability is better than 10 μV/week.

(Bailey Meters & Controls, Ltd., Purley Way, Croydon.)

1368

Right: Graphite for Hinkley Point, showing ease of handling and stacking with the Reed standard packages.



Packaging Graphite

Nuclear engineers need no reminding of the problems which may be faced in the storage and transport of graphite moderator blocks—the need for almost clinical cleanliness, in addition to protection from damage. An acceptable solution is provided, in the case of Hinkley Point power station, by Reed Corrugated Cases, Ltd., who have produced a heavy-duty corrugated container for the purpose. This has been designed so that a single outer case will accommodate all variations of block used in the pile, different sets of corrugated fibreboard inner fittings being used as necessary. It is scarcely necessary to emphasize the simplification in stacking and handling achieved by a standard pack.

(Albert E. Reed and Co., Ltd., Packing Division, Reed House, Kew Bridge, Brentford, Middlesex.)

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CATALOGUES

Laboratory muffles and small furnaces; shaker hearth furnaces; induction heating equipment; gas carburizing and nitriding processes are the subject of new publicity material received from **Wild-Barfield Electric Furnaces, Ltd.,** Elecfurn Works, Otterspool Way, Watford By-pass, Herts.

"A.C. to D.C. Transistorised Converters" by R. A. Holmes, is an application report recently published by **Standard Telephones and Cables, Ltd.** (Transistor Division), Footscray, Kent.

The Computer Department of **Ferranti, Ltd.,** 21 Portland Place, London, W.1, have issued a classified index of computer literature.

Environmental testing and apparatus for controlled atmospheric conditions is the subject of a 32-page catalogue received from **Webber Manufacturing Co., Inc.,** P.O. Box 217, Indianapolis 6, Ind., U.S.A.

New lists from **Guest Keen and Nettlefolds (Midlands), Ltd.,** Box No. 24, Heath Street, Birmingham, 18, cover special fastening equipment, including self-tapping cladding screws, lightweight nuts, hollow steel and ultra high tensile steel bolts and titanium alloy bolts.

Epoxy resins for the electrical industry are the subject of Technical Note 211 published by **CIBA (A.R.L.), Ltd.,** Duxford, Cambridge. Another publication received from the same source is "Introduction to Duxford and to the products of CIBA" giving a general description of the manufacturing plant for Araldite and associated products.

A bulletin from **Fibrous Glass Products Inc.,** a subsidiary of Pall Corporation, Alpa Plaza, Hicksville, Long Island, N.Y., gives technical data and applications of glass fibre.

New catalogues from **Bailey Meters and Controls, Ltd.,** Purley Way, Croydon, include the "Bailetronic" process control, pneumatic control drives, the "Mini-Line" level transmitter and their proportional pneumatic computing relay.

Vacuum induction furnaces and vacuum resistor furnaces are the subject of a new publication received from the **General Electric Co., Ltd.,** Erith, Kent.

Precious metal contacts of all types are described in a 48-page brochure received from **Engelhard Industries, Ltd.** (Baker Platinum Division), 52 High Holborn, London, W.C.1.

Wallace abrasive cut machining is described in a leaflet received from **B. O. Morris, Ltd.,** Morrisflex Works, Briton Road, Coventry.

Vol. 3, No. 2 of the *Fairey Review* (**The Fairey Co., Ltd.,** 24 Bruton Street, London, W.1) includes a table of the elements and their isotopes as a supplement to an article, "A Plain Man's Guide to Atomic Energy."

Skatoskalo scaling machines for all types of tube and surface scaling problems are described in a catalogue received from **Gilmans Flexible Drives, Ltd.,** Millers Road, Warwick.

A booklet describing demineralization plant has been received from **Head Wrighton and Co., Ltd.,** Ship House, 20 Buckingham Gate, London, S.W.1, who have acquired the British rights of Belco (U.S.A.) water treatment.

A maintenance welding data book listing applications of 150 different low-temperature welding alloys and fluxes, with notes on technique and joint design has been received from **Eutectic Welding Alloys Co., Ltd.,** North Feltham Trading Estate, Feltham, Middlesex.

"Synchro Accessories and Mounting Instructions" is the title of a new publication received from **Muirhead and Co., Ltd.,** Beckenham, Kent, giving hints on mounting all types of servo components.

The range of M and I glass-based insulating materials with full technical data is described in a publication received from the **Micanite and Insulators Co., Ltd.,** Empire Works, Blackhorse Lane, Walthamstow, London, E.17.

"Birlec Equipment in Industry" and "Annealing Furnaces for Malleable Iron" are two new publications received from **AEI-Birlec, Ltd.,** Tyburn Road, Erdington, Brimingham, 24.

New leaflets from **Miles Electronics, Ltd.,** Shoreham Airport, Sussex, cover coefficient units and integrator units for computers.

Heat treatment furnaces of all types are described in a publication received from the **Electric Resistance Furnace Co., Ltd.,** Netherby, Queens Road, Weybridge, Surrey.

Control gear for machine tools is covered by a new catalogue received from **Allen West and Co., Ltd.,** Brighton, 7.

Data handling systems by line or V.H.F. radio are described in a publication received from **Sir W. G. Armstrong Whitworth Aircraft, Ltd.,** Baginton, Coventry.

From **James Gordon and Co., Ltd.,** Dalston Gardens, Stanmore, Middlesex, come two leaflets describing the B.t.u. meter, and distance boiler water level indication.

Patents Reviewed

These abstracts have been made from British Patent Specifications, complete copies of which can be obtained from the Patent Office, 25 Southampton Street, London, W.C.2, at 3s. 6d. each (including postage).

B.P. 831,226. Electromagnetic flowmeter. J. A. Shercliff. To: U.K. Atomic Energy Authority.

A rotatable spool whose cheeks carry equidistant magnetic poles, each facing a pole of opposite polarity on the other cheek. Liquid metal in a tube between the cheeks causes the spool to rotate by eddy current drag, its rotational speed being a measure of the rate of flow of the liquid metal.

B.P. 831,323. Handling plant for spent nuclear reactor fuel elements. R. A. Taylor. To: Babcock and Wilcox, Ltd.

A container with receiving spaces for the fuel elements is moved by a rotor to a series of positions in which a space is opposite a fuel element discharge aperture of the discharging machine. The container is then moved to a location below the water of a storage pond.

B.P. 831,383. Liquid-liquid contacting apparatus. M. C. Tanner, L. Lowes. To: U.K. Atomic Energy Authority.

The mixer-settler apparatus for the contacting of toxic, radioactive or otherwise dangerous liquid has pairs of mixer tubes dipping into each mixer compartment. The pressure in the mixer tubes can be increased and decreased in and out of phase relationship by a double-acting piston and cylinder arrangement.

B.P. 831,433. Fuel elements for nuclear reactors. E. R. Elliott, D. C. Brown. To: C. A. Parsons and Co., Ltd.

The container housing the fuel extends beyond the fuel at each end and is closed by a plug which fits into the space between the fuel and the end of the container while a portion of the plug extends beyond the end of the container abutting the end face of the container. These abutting faces are welded together. The outer plug ends have either a projecting cone or a conical recess to locate the respective adjacent fuel elements in the same channel. Lugs on the plugs facilitate the lifting of the elements.

B.P. 831,547. Graphite moderated units. I. A. B. Gaunt. To: General Electric Co., Ltd.

This refers to graphite blocks to form columns with a fuel channel through which a coolant (e.g., carbon dioxide) flows. The blocks should be so designed as to keep leakage in the column to a minimum, and they should be accurate and easily located in position with minimum distortion of the locating arrangement due to Wigner growth. The blocks are provided with upper and lower ribs, e.g., along opposite sides of a rectangular block, and corresponding recesses along the other sides of the block, so that the projecting ribs of one block form keys to fit into the recesses of the adjacent blocks. The projections have their major axis in the direction of the grain of the graphite while the recesses have their major axis at right angles to the grain of the graphite.

B.P. 831,679. Ceramic nuclear fuel element. To: Norton Grinding Wheel Co., Ltd. (U.S.A.).

A core composed of a refractory oxide, and a material selected from the group of fissile and fertile material and mixtures, in a sheath integral with the core, the sheath being predominantly alumina. Such elements can be simply stacked in a nuclear reactor and coolant (water, helium) can be passed through without the provision of separate conduits.

B.P. 831,722. Nuclear fuel element handling apparatus. J. M. Nicol. To: Babcock and Wilcox, Ltd.

When fuel elements are withdrawn upwards from an atomic pile by a discharge machine, they are usually lowered from the machine in an upright position, and transported in a container in this upright position to a pond. This position makes it difficult to make best use of the space in the container. In the new apparatus the holder for the fuel elements is movable between an upright receiving position and a discharge position in which it lies over the container. The pivotally mounted holder may have doors, which form its lower part in the discharging position, to be opened by a cam arrangement during the end part of the movement over the discharge station. Or the holder may incorporate a rotatable member for the fuel element which is turned round to let the element drop through an aperture in the container. This aperture is closed by the rotatable member in the holding position.

B.P. 831,861. Neutron responsive device. To: U.K. Atomic Energy Authority (U.S.A.).

A bi-metallic strip composed of two metallic strips having different coefficients of expansion are secured together. One of the strips contains fissile material so that, when exposed to neutron sources, it causes heating and consequent flexure of the bi-metallic strip. This flexure can be used to open and close a switch.

B.P. 832,270. Thermonuclear reactor. J. E. Allen, R. J. Bickerton. To: U.K. Atomic Energy Authority.

Refers to reactors in which a high current ring discharge is induced in a gas contained in a torus (B.P. 830,252). The conducting toroid is divided into a number of sectors electrically insulated from each other, each sector being fed simultaneously with a current pulse from a single source (condenser) or each from a separate source (charged transmission line).

B.P. 832,366. Control systems for nuclear reactors. P. F. Hartshorn. To: General Electric Co., Ltd.

The controlling means (rods of neutron-absorbing material) are actuated in response to an error voltage derived from the difference between a voltage representing existing reactor power and a feedback voltage obtained from an adjustable tapping on a potentiometer, the potential difference across whose ends depends upon the sum of the error voltage and a reference voltage. To minimize oscillation of the power level a stabilizing network is preferably included. The voltage difference may be amplified, after being applied across a d.c.-a.c. converter, to give the error voltage to actuate the control means through at least one relay. The voltage representing the existing reactor power is obtained from a resistance through which current flows from an ion chamber.

B.P. 832,367. Winding and haulage equipment. D. S. Mofrey. To: British Thomson-Houston Co., Ltd.

A free-running pulley, around which the rope or cable for raising or lowering a control rod passes, is mounted in a pivotally arranged carrier. If the pressure of the rope turns the carrier against spring pressure about a certain angle in one direction a switch is actuated. If due to a decrease of rope tension the carrier turns a certain amount in the opposite direction a second switch is actuated. These switches control the rotation of the winding motor.

B.P. 832,506. Process for the treatment of an aqueous homogeneous nuclear reactor fuel. E. Glueckauf. To: U.K. Atomic Energy Authority.

The rate of removal of fission products from the fuel (a homogeneous aqueous solution of uranyl sulphate) is increased by the (continuous) addition of a metal sulphate (cerium sulphate, barium sulphate, silver sulphate) or a gas (iodine).

B.P. 832,694. Mechanisms for controlling the operation of nuclear reactors. D. M. Sutherland, H. Proctor, J. K. Jobbins, J. L. Kearney. To: Metropolitan-Vickers Electrical Co., Ltd.

In an emergency shut-down of a nuclear reactor a neutron absorbing mass has to be moved as quickly as possible over most of its defined range, then arrested in the remainder of its range of travel without undue stress. This is achieved by an eddy-current brake connected with an eddy-current rotor in a magnetic field set up by permanent magnets.

B.P. 832,877. Power plant for nuclear reactors. To: Babcock and Wilcox, Ltd. (France).

It is difficult to reheat steam between different pressure stages of the turbine, particularly before the low pressure stage, because of the distances required for biological protection. The problem has been solved by the arrangement of an economizer heated by the fluid heat carrier and supplying heated liquid to the heat exchanger for evaporation and a reheater adjacent to the turbine, together with means for receiving part of the heated liquid from the economizer and to transfer heat from that liquid to the steam between stages.

B.P. 832,959. Silver alloy for the control rods of nuclear reactors. To: U.S. Atomic Energy Commission (U.S.A.).

Addition of a noble metal (platinum, ruthenium, rhodium, palladium) in amounts from 0.5 to 1.5 (weight) % to a cadmium-indium-silver alloy imparts superior structural strength properties.

B.P. 833,094. Moderator or reflector substances for nuclear reactors. To: Siemens-Schuckertwerke A.G (Germany).

Solutions of lithium and/or beryllium compounds in light and/or heavy water (Li^7OD or BeF_2 in heavy water, LiOH or BeF_2 in light water).

B.P. 833,332. Self-lubricating bearings. A. Blainey. To: U.K. Atomic Energy Authority.

A suspension of particles of zirconium in an evaporable medium is applied to the cleaned metal surface and heated to at least 900°C to bond the particles to the metal as a porous layer. The layer is then impregnated with polytetrafluoroethylene.